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POWER REACTOR TECHNOLOGY

A Quarterly Technical Progress Review

Prepared for U. S. ATOMIC ENERGY COMMISSION by GENERAL NUCLEAR ENGINEERING CORP.

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● VOLUME 3

● NUMBER 2

TECHNICAL PROGRESS REVIEWS

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POWER REACTOR TECHNOLOGY

A REVIEW OF RECENT DEVELOPMENTS

Prepared for U. S. ATOMIC ENERGY COMMISSION
by GENERAL NUCLEAR ENGINEERING CORP.

● MARCH 1960

● VOLUME 3

● NUMBER 2

foreword

This quarterly review of reactor development has been prepared at the request of the Division of Information Services of the U. S. Atomic Energy Commission. Its purpose is to assist interested organizations in the task of keeping abreast of new results in reactor technology for civilian application.

The report is a concise discussion of selected phases of research and development for which there have been significant advances or a heightened interest in the past few months. It is not meant to be a comprehensive abstract of all material published during the quarter, nor is it meant to be a treatise on any part of the subject. The intention is to cover the various areas of reactor development from the general viewpoint of the reactor designer rather than from the more detailed points of view of specialists in the individual areas. However, papers which are thought to be of particular significance or particular usefulness in specialized fields will be mentioned in short notes. In the over-all plan of the report, it is intended that various subjects will be treated from time to time and will be brought up to date at that time.

Any interpretation of results which is given represents only the opinion of the editors of the report, who are General Nuclear Engineering Corporation personnel. Readers are urged to consult the original references in order to obtain all the background of the work reported and to obtain the interpretation of the results given by the original authors.

W. H. ZINN
General Nuclear Engineering Corporation

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GENERAL RESEARCH AND DEVELOPMENT

EVALUATIONS

Large Power Reactors

At the November meeting of the Atomic Industrial Forum, Frank K. Pittman, Director of the Division of Reactor Development of the Atomic Energy Commission, summarized the results of a study[1] carried out by the AEC, with the assistance of individuals from the national laboratories and from industry, to evaluate the potentials of several reactor types for the generation of economic nuclear power. The study is part of a project to review and evaluate the results of the first five years of the U. S. nuclear power program and to determine the course to be pursued during the next phase of the program. This project has generally been referred to as the "Ten-year Program."

The results of the evaluation study, as presented by Mr. Pittman, are summarized in Table I-1, in the form of estimated power-generation costs for the several reactor types. Two cases are treated for each reactor type: the first is considered representative of the best existing technology, and the second is considered possible at a future date, provided that the research and development programs which can presently be defined specifically are carried out and yield successful results. Also listed in the table are the estimated dates upon which construction of the potential plants could be started, the costs of the research and development programs involved, and the number and cost of prototypes judged necessary before start of construction of the potential plant.

In Table I-2 the ground rules used in making the cost estimates are summarized. Besides these specific ground rules, the general principles followed were: all estimates were based on single-reactor single-turbine stations of 300 Mw(e) net output, and no specific allowance was made for cost reductions that might result primarily from an assumed larger volume of nuclear power business than that which presently exists. In Table I-3 some of the characteristics which are important in determining power costs are summarized for the several reactor types. The important areas in which improvements are needed, and in which the major cost reductions are sought, are summarized below for each reactor type.

Pressurized-water Systems. These systems are considered the most advanced technologically. The capital-cost reductions for the "potential plant" are expected to result from the shift from canned rotor pumps to pumps with mechanical shaft seals, the elimination of external pressurizers, the modifications of vapor containment, the use of bulk boiling, and the simplification and improvement of control drives. At least two of these developments (bulk boiling and elimination of external pressurizers) are in the direction of a closer approach to the boiling reactor. Fuel-cost reductions are expected to result from (1) an increase in the maximum fuel exposure from 27,500 to 40,000 Mwd/metric ton and (2) a decrease in the cost of fuel fabrication (Table I-3). It is not clear whether the expected increase in permissible fuel exposure depends upon a real development in UO_2 fuel-element technology or simply upon extended irradiation testing to prove that current fuel-element types can, in fact, tolerate such exposures. It is to be noted that the 13,000 Mwd/metric ton average exposure expected for the plant built with "today's technology" implies a maximum/average exposure ratio of only 2.11. Evidently a partial fuel reloading scheme with some sort of fuel-element shuffling is visualized.

Boiling-water Systems. The boiling-water systems are considered only slightly less advanced technologically than the pressurized-

Table I-1 ECONOMIC POTENTIAL OF VARIOUS REACTOR TYPES[1,2]

	Enriched thermal converters: objective is competitive power as soon as possible						Natural U reactors: objective is operation on natural U feed	Breeders: objective is breeding with a reasonable doubling time	
	PWR	BWR	BWR with nuclear superheat	OMR	SGR	GCR	D_2O natural U	Fast breeder	Aqueous homogeneous
Today's technology:									
Capital cost, mills/kw-hr	5.05	5.26		4.39	6.11	5.97	7.05	5.10	
Fuel cost, mills/kw-hr	3.38	3.47		5.72	4.12	3.21	4.22	7.10	
Operating cost, mills/kw-hr	0.85	0.88		1.34	0.99	1.18	1.23	1.05	
Total	9.28	9.61		11.45	11.22	10.36	12.50	13.25	
Potential plant:									
Capital cost, mills/kw-hr	4.40	4.31	3.88	3.65	4.92	4.09	6.49	4.43	6.38
Fuel cost, mills/kw-hr	2.56	2.29	1.99	2.11	1.92	2.80	1.31	1.99	2.12
Operating cost, mills/kw-hr	0.85	0.88	0.88	1.34	0.94	0.78	1.23	1.05	2.83
Total	7.81	7.48	6.75	7.10	7.78	7.67	9.03	7.47	11.33
Cost reduction:									
Capital cost, mills/kw-hr	0.65 = 13%	0.95 = 18%	0.43 = 10%*	0.74 = 17%	1.19 = 19%	1.88 = 31%	0.56 = 8%	0.67 = 13%	
Fuel cost, mills/kw-hr	0.82 = 24%	1.18 = 34%	0.30 = 13%*	3.61 = 63%	2.20 = 53%	0.41 = 13%	2.91 = 69%	5.11 = 72%	
Operating cost, mills/kw-hr	0	0	0	0	0.05 = 5%	0.40 = 34%	0	0	
Total	1.47 = 16%	2.13 = 22%	0.73 = 10%*	4.35 = 38%	3.44 = 31%	2.69 = 26%	3.47 = 28%	5.78 = 44%	
Possible date of construction start on potential plant	Near future	1964-1965	1965	1965	1965-1966	1966	1967-1968	1968-1969	1970-1971
Cost of research and development before potential plant, $10^6	20	20	4	40	82	100	80	160	75
Prototypes needed before potential plant	None	High-power-density prototype	1 prototype	75-Mw(e) prototype	75-Mw(e) prototype beyond Hallam	Oak Ridge prototype (under construction)	Prototypes beyond Carolinas-Virginia	1 prototype beyond Fermi	2 experimental reactors; 1 protype [125 Mw(e)]
Cost of prototypes, $10^6	0	25	20	30	50	30	175	55	100

*Reduction below potential BWR costs.

Table I-2 GROUND RULES FOR COST ESTIMATES[2]

Capital Costs and Charges

General and administrative expenses: 12.5% of direct cost
Engineering, design, and inspection: 14.6% of direct cost plus general and administrative expenses
Start-up cost: based on 4½-month operating cost
Contingencies: 10% of direct and indirect costs
Interest during construction (total): 8.1% of total cost
Capital charge rate: 14% per annum
Load factor: 80%

Fuel-cycle Costs

Uranium costs: AEC schedule for UF_6
Plutonium value, $12 per gram
Chemical reprocessing cost: $15,300 (8+ metric tons of U)
Shipping cost of irradiated fuel: $12.45 per kilogram
Conversion of uranyl nitrate to UF_6: $5.60 per kilogram
Conversion of plutonium nitrate to Pu: $1.50 per kilogram
Fuel use charge: 4% per annum
Capital charge on working capital: 6% per annum

Operation, Maintenance, and Insurance Costs

Operation and maintenance costs: based on labor estimates, supply estimates, and allowance for extraordinary items
All-risk nuclear property insurance: $0.35 per $100
Nuclear liability insurance from insurance pool, $150,000 of insurance per megawatt (thermal) at the following rates:

	Amount	Rate/million	Annual premium
First	$ 1,000,000	$40,000	$ 40,000
Next	4,000,000	20,000	80,000
Next	5,000,000	8,000	40,000
Next	10,000,000	4,000	40,000
Next	20,000,000	2,000	40,000
Next	20,000,000	1,000	20,000
	$60,000,000		$260,000

Government indemnity insurance: $30 per megawatt (thermal) for a maximum coverage of $500,000,000

water systems. The today's technology estimates are based on a dual-cycle plant, which evidently is not considered to be the economic optimum. It is not clear why a single-cycle plant should be considered outside the realm of existing technology. In any event the capital-cost reductions of the potential plant are expected to result primarily from the elimination of large risers and heat exchangers, an increase in power density from 28 to 50 kw per liter of core, and changes in the vapor containment. Fuel-cycle costs are expected to be reduced by power flattening, by increasing the average irradiation of the fuel to 19,000 Mwd/metric ton (presumably as a result of power flattening plus an extension of the maximum exposure to 40,000 Mwd/metric ton), and by reduction of the fabrication cost. The average fuel exposure for the today's technology plant is 11,000 Mwd/metric ton (Table I-3), apparently reflecting a belief that the maximum/average exposure ratios are currently higher for boiling-water than for pressurized-water reactors. It is also to be noted that the boiling reactor uses Zircaloy fuel jackets, whereas the pressurized-water reactor uses stainless steel.

Nuclear Superheat. Today's technology is not considered adequate for a large boiling reactor with nuclear superheat; consequently only the "potential plant" is listed in the table. The power-cost reductions expected beyond those of the "potential" boiling-water plant result from higher plant efficiency and lower turbine cost.

Organic-cooled and -moderated Systems. The present deficiencies of these systems are considered to be mainly the high fuel-cycle costs, which result from the low exposure limits of the metal fuel elements used in the present organic reactors, and the replacement cost of radiation-damaged organic material. The cost reductions for the potential plant are expected to result mainly from the development of a new fuel element and cladding material, an increase in power density from 19 to 44 kw per liter of core, and an increase in the heat-transfer capabilities of the system. The type of fuel element anticipated is not indicated.

Sodium Graphite Systems. The lack of high-exposure fuel elements is considered the outstanding deficiency of current sodium graphite systems also. The fuel-cycle cost reductions for the potential plant are expected to result mainly from the development of a new fuel element, probably uranium carbide, with a life of 17,000 Mwd/metric ton, and from reduction of the fuel-fabrication cost. Capital-cost reductions are expected from the development of simplified and much less expensive steam generators, the confining of the primary cooling system to the reactor vessel, the elimination of moving parts in the sodium stream, and the development of a calandria type core to supersede current canned-moderator cores.

Gas-cooled Systems. Only gas-cooled systems using enriched fuel were studied. For these systems the major capital-cost reductions are expected to result from increasing the bulk coolant outlet temperature from 1050 to 1200°F, increasing the system pressure from 300 to 400

Table I-3 SIGNIFICANT CHARACTERISTICS OF VARIOUS REACTORS[1,2] USED IN POWER-COST ESTIMATES

	PWR	BWR	BWR with nuclear superheat	OMR	SGR	GCR	D_2O natural U	Fast breeder	Aqueous homogeneous
Capital cost, dollars/kw(e):									
Today's technology	252	263		220	306	298	360	255	
Potential plant	220	215	194	183	246	204	331	221	318
Fuel-cycle cost, mills/kw-hr:									
Today's technology	3.38	3.47		5.72	4.12	3.21	4.22	7.10	
Potential plant	2.56	2.29	1.99	2.11	1.92	2.80	1.31	1.99	2.12
Fuel material:									
Today's technology	UO_2 in S.S.	UO_2 in Zircaloy		U-Mo in Al	U-Mo in S.S.	UO_2 in S.S.	U metal in Zircaloy	U-Mo in S.S.	
Potential plant	UO_2 in S.S.	UO_2 in Zircaloy		New type	UC	UO_2 in S.S.	U metal in Zircaloy	Cermet	
Maximum fuel exposure, Mwd/metric ton:									
Today's technology	27,500	27,500							
Potential plant	40,000	40,000							
Average fuel exposure, Mwd/metric ton:									
Today's technology	13,000	11,000		4500	3000	10,000	3850	15,900	
Potential plant		19,000			17,000	14,500	7000		
Fuel-fabrication cost, dollars per kg of U:									
Today's technology	110	140		60	80–110	110	50	480	
Potential plant	70	90			70	70	15		

psi, and increasing the power density from 5 to 8 kw per liter of core. Fuel costs are expected to be lowered by extending fuel life from 10,000 to 14,500 Mwd/metric ton and by decreasing fabrication costs. Operating-cost reductions would result from substituting CO_2 for helium as coolant. This, of course, would be a major change in the system.

D_2O-moderated Natural-uranium Systems. The natural-uranium D_2O reactor considered was a pressure-vessel nonboiling type using metal fuel. Capital-cost reductions are expected to result from a change to a pressure-tube design and from an increase of power density, from approximately 26 to 35 kw/liter. Fuel-cost reductions would result from an increase in exposure lifetime to 7000 Mwd/metric ton and a decrease of fabrication cost from $50 to $15 per kilogram of uranium. It is evidently contemplated that the uranium-metal fuel elements will be retained. The very low fabrication cost anticipated reflects the developmental approach which is being pursued at Savannah River (see the section on Heavy-water Reactors in this issue).

Fast Reactor Systems. Here again the lack of a suitable fuel element is considered to be the major problem. The shift from a uranium–10 wt.% molybdenum element to a cermet element of unspecified composition is counted on for a very large reduction in the fuel cost (from 7.10 to 1.99 mills/kw-hr). Capital cost is expected to be lowered by reduction of the reactor shield thickness, simplification of piping layout and of the fuel loading and unloading mechanism, increase of reactor temperature rise, decrease of temperature drop through the intermediate heat exchanger and steam generator, decrease of pump and pipe sizes, and improvement of instrumentation.

Aqueous Homogeneous Systems. These systems are directed toward breeding in the thermal-neutron spectrum. It is considered that their technical feasibility as thermal breeders has not yet been demonstrated, and therefore only the potential plant appears in the table.

If the cost estimates are considered on a comparative basis, there are at least seven reactor types which must be considered as potentially competitive with fossil-fuel plants in the rea-

sonably near future. The number might have been extended to eight if an enriched version of the D_2O reactor had been studied. It must, of course, be remembered that the potential plant estimates are based on the assumption that all the anticipated research and development are successful, and the magnitude of these programs varies over quite a wide range, perhaps more than is reflected in the range of possible starting dates for the potential plants of the various types. The probability of successful accomplishment would be expected to increase as the magnitude of the development effort decreases.

If the magnitudes of the estimated power costs are accepted, then the future for competitive nuclear power looks very encouraging despite the uncertainty which exists because of the need for further research and development. Surely it is reasonable to expect that the research and development programs will be successful in at least one case out of the seven. The estimates are, of course, subject to other uncertainties. The impression, for example, is that the potential plant estimates were made by estimating, first, the power cost for the today's technology plants and applying to these estimates the cost savings expected as a result of the successful research and development. Although this method has much to recommend it, it does present opportunities for additive errors as well as the temptation to recognize the benefits of new developments without recognizing problems which may accompany them. Finally, it is worth noting that in all cases except one (the gas-cooled reactor) the largest power-cost decreases in going to the potential plant result from reductions in the fuel-cycle costs. It is difficult to define what is meant by "competitive power" when all the major components of the fuel-cycle cost except one (the fabrication cost) depend upon government-fixed rates.

That the uncertainties are recognized by the Commission is evident from Mr. Pittman's remarks in his address:[1]

> In closing, I would like to emphasize once more that the studies which I have reported on here today represent only one small part of the continuing evaluation and analysis program which will play an ever increasing part in developing our national nuclear energy program. The figures I have given and the conclusions I draw today are as important for what they leave unsaid as for what they say. The figures have all the uncertainties that are inherent in any predictions based on the results of experimental programs yet to be undertaken and on the operation of plants not yet built, or in many cases not yet even designed. In developing the figures, we were naturally forced to use a fair share of technical and scientific judgment. I can only hope and request that an equal degree of understanding and judgment is exercised by those who will use and quote the figures.

Presumably the Division of Reactor Development itself will be the only direct user of the figures, and a good deal more interest attaches to the use that will be made of them than to the figures themselves, particularly since it has been said that the estimates are to play a part in determining the AEC program of reactor development. The significance of the estimates can be understood more fully in relation to the stated objectives of the program, which have also been evolved in connection with the Ten-year Program study. The objectives are as follows:[2]

1. The achievement of nuclear-power-generation costs that are comparative with power costs of fossil-fueled plants, on an equivalent station-design basis, in high-energy-cost areas in this country, by 1968. High-energy-cost areas are defined as those where the cost of fossil fuel is $0.35 per million Btu (on 1959 cost).

2. The achievement of nuclear-power-generation costs that are comparative with fossil-fueled plant costs, on an equivalent station-design basis, in all but the low-energy-cost areas, by 1975. Low-energy-cost areas are defined as those where the cost of fossil fuel is $0.25 per million Btu or less (on 1959 cost).

3. The successful development of (*a*) breeder type reactors that will serve as sources of plutonium and U^{233}, with primary emphasis on the best breeding potential, and (*b*) a breeder system that will also produce competitive power.

4. The successful development of a reactor type whose operation is independent of a continuous feed of enriched fuel: natural-uranium reactors or self-sufficient recycle reactors.

According to the statements made in connection with the program study, it is intended that these objectives, and the analyses which attempt to determine which reactor systems can best meet the objectives, shall carry real weight in determining the program of the AEC. The following quotation is significant:[2] "It may also be impossible to accurately determine the one concept with the most potential (for meeting one of

the objectives) at this status of the technology. There is no reason, however, to continue the development of those concepts competing for the same objective that do not show the same degree of long-range potential."

Certainly it would be hard to say that the stated objectives are anything but desirable ones, or that development should be continued on reactors which can be shown to be less capable than other types in meeting a set of adequate objectives. There could be some question as to whether the stated objectives define an adequate program and a further question as to whether the separation of related characteristics into separate specific objectives — however desirable this may be from the standpoint of administering a vigorous program — may not distort the relations. Some evidences of a distortion of the latter type may be apparent in the results of the evaluation study. The various reactor types have been grouped according to the specific objectives for which they apparently were considered in the study, and this grouping has been indicated, in Table I-1. The enriched thermal converters, for example, were evidently considered specifically for meeting objective No. 1, and perhaps No. 2. One wonders whether, within this group of enriched thermal converters, the question of neutron economy per se is to be ignored and whether, for example, the H_2O reactor which employs zirconium as structural material and achieves a reasonably high conversion ratio is to have no advantage over one which employs steel as structural material and has a low conversion ratio. Surely, if the concern for the long-term resources of nuclear fuel is sufficient to justify an expensive development program on breeder reactors, and if the concern for the shorter term resources is sufficient to justify a natural-uranium program, then there must be some concern for the efficiency of fuel utilization in those reactors which first will become competitive. In a similar vein, one wonders whether it is to be recognized that a reactor which could conceivably meet two objectives (for example, the D_2O reactors which in their enriched versions might have a good potential for early competitive power generation in addition to their potential for natural-uranium operation) is more desirable than reactors which inherently are limited to meeting a single objective.

Questions and speculations such as these will naturally occur during the interim period while the results of the program study are still incomplete. The complete results, and whatever actions or recommendations of action may result from them, will of course be awaited with interest.

Small Power Reactors

By a memorandum dated Mar. 10, 1959, the AEC assigned to the Oak Ridge Operations Office the responsibility for formulating and implementing a program for consideration of small-sized nuclear power plants (SSNPP). The immediate questions considered were: What specific size and type of nuclear reactor system would be feasible for immediate construction under this program, considering both the present state of the art and the potential for significant improvements in performance? What is the status of such a nuclear plant at the present time in comparison with a conventional plant located in a high-cost fuel area?

By legislative direction the AEC would design, construct, and own the reactor, with the site and conventional generating equipment to be furnished by a cooperative or publicly owned power-generating organization. The AEC would then sell steam to the organization at rates based upon the present or projected cost of comparable steam from a conventional plant. Initial screening of various reactor types indicated that only the pressurized-water (PWR), boiling-water (BWR), and organic-moderated (OMR) reactors should receive detailed study during the first phase of the program. Accordingly, contracts were let with Alco Products, Inc., General Nuclear Engineering Corporation, and Atomics International to evaluate the respective reactor types. It was established that fossil-fuel superheat was economically attractive, and it was accordingly considered in the evaluation of the three concepts.

To compare the economics of existing small conventional power plants and proposed small nuclear power plants, a survey was made of existing U. S. conventional generating plants from 5- to 44-Mw(e) capacity.[3] A questionnaire was mailed to 217 cooperatives and publicly owned organizations having generating capacities in excess of 500 kw. The results of the questionnaire are quoted as follows:[4]

1. The questionnaire was taken quite seriously by the majority of recipients as attested by the fact that 73.8 per cent filled out and returned the form.

2. Of those returning the questionnaire, 71.5 per cent indicated they would be interested in planning an SSNPP to supplement their generating capabilities if AEC instituted a program for a joint arrangement.

3. Out of 186 responding organizations, 58 had added a plant or unit of 5- to 44-Mw capacity in the last five years, and 72 organizations were constructing or considering constructing a plant or unit in this range during the next two years.

4. The annual capacity factor of the generating units or plants added in the last five years averaged 77 per cent for cooperatives and 61 per cent for municipalities. These high capacity factors are apparently due to the fact that the new units are generally used for the base load. In contrast, the Rural Electrification Administration (REA) reports that 26 REA-financed generating plants had an over-all plant factor of 56.1 per cent. A number of the units in the REA tabulation were 17, 23, or 31 years old and were probably used principally for peaking purposes.

5. The sizes and numbers of each size of generating units built or being planned are indicated below for the range of interest to the SSNPP program:

	Built	Planned	Total
16.5-Mw gross	7	11	18
20.0-Mw gross	2	3	5
22.0-Mw gross	9	10	19
22.5-Mw gross		1	1
25.0-Mw gross		2	2

6. Of the 130 cooperatives and municipalities submitting data on units added or planned, it was found that all units built in the last five years use superheat. All units being constructed or planned in the next two years will also use superheat. There were no exceptions to this situation in the capacity range of 5 to 44 Mw. This is considered significant as it indicates the unanimous desire of publicly owned organizations to use superheat to attain high thermal efficiency. It is noted that the public power organizations planning new generating units in the size range of 16.5 Mw through 25.0 Mw are stipulating steam pressures varying between 600 and 900 psig with temperatures ranging between 750 and 1000°F.

"High-cost" fossil fuel was found to have an average cost of $0.50 per 10^6 Btu; the average capital cost of a 22.0-Mw(e) plant built in the last five years was $173 per kilowatt, whereas for a plant being built or planned for the next two years the capital cost was $213 per kilowatt. For a fuel cost of $0.50 per 10^6 Btu, the fuel cost of the 22.0-Mw(e) (gross) plant was reported to be 6.4 mills/kw-hr. Thus the 22-Mw(e) conventional plant generates electricity for about 10 mills/kw-hr in a high-cost fuel area. For the average municipality operating at a 61 per cent load factor, the fixed annual charge was approximately 5.7 per cent of the initial investment.

A discussion of the studies of the three reactor types follows.

Boiling-water Reactor (BWR). Reference 5 reports on the potential of the BWR with respect to small generating plants. After summarizing data on completed and planned boiling-water reactors, the publication presents two reference designs, one for a boiling reactor without superheat and one for a boiling reactor with oil-fired superheat. Since the relatively lower power cost of the superheat reactor makes it look more attractive than the nonsuperheat reactor, the latter will not be discussed.

The characteristics of the reference design are given in Table I-4. The reactor is of the direct-cycle natural-circulation type. The operating pressure of 1450 psig is considerably greater than that of any operating BWR, and it is expected to be an advantage from a stability viewpoint. One possible problem resulting from the combination of a direct-cycle reactor and an oil-fired superheater is the effect of the high oxygen content of the steam from the reactor on the corrosion rate of the superheater tubes. This possibility was regarded as a tolerable uncertainty, the resolution of which would be an added incentive for building and operating the reactor.

Organic-moderated Reactor (OMR). The OMR was discussed in two volumes. Reference 6 presents general information on the present and future OMR plants, the performance characteristics of uranium alloy fuels, and the basic economics of organic-moderated reactors. Reference 7 presents the reference design for a 20-Mw(e) OMR without fossil-fuel superheat; the cost details of the reactor with fossil-fuel superheat were supplied verbally by Atomics International representatives at a meeting with the AEC in May 1959.

Table I-5 contains pertinent data on the 20-Mw(e) organic-moderated power plant. The reactor is more conservative in design than the marine propulsion reactor discussed in a following section of this Review, in that neither

Table I-4 DESIGN CHARACTERISTICS OF BOILING REACTOR WITH OIL-FIRED SUPERHEAT

[Plant Rating: 25 Mw(e)]

General	
Gross reactor thermal power, Mw	47.0
Net reactor thermal power, Mw	46.0
Gross superheater thermal power, Mw	24.7
Net superheater thermal power, Mw	19.8
Gross electric output, kw	25,000
Gross cycle efficiency, %	38.1
Gross plant efficiency, %	34.9
Net electric output, kw	~23,500
Reactor operating pressure, psig	1450
Temperature of steam leaving reactor, °F	593
Pressure of steam at turbine, psig	1250
Temperature of steam at turbine, °F	1000
Total steam flow, lb/hr	206,000
Feed-water temperature, °F	430
Coolant circulation	Natural
Pressure vessel ID, ft	6.71
Pressure-vessel wall thickness, in.	4.8
No. of turbines	1
Fuel Elements	
Diameter of UO_2 pellets, in.	0.450
Cladding material	Zircaloy-2
Cladding thickness, in.	0.027
Diameter of clad rods, in.	0.510
No. of rods per assembly	25
Assembly structural material	Zircaloy-2
Assembly outside dimensions, in.	4.0 × 4.0
Assembly wall thickness, in.	0.060
Core lattice, sq in.	4.5
No. of assemblies in core	124
Reactor Core	
Control method	13 vertical cruciform rods, bottom driven
Active height, ft	4.33
Effective diameter, ft	4.71
Fuel loading, tons of U	4.19
Thermal power, Mw	47.0
Average specific power, Mw per metric ton of U	11.23
Power density in coolant, kw/liter	44.7
U^{235} content of fuel, %	1.7
Exit coolant temperature, °F	593
Maximum cladding-surface temperature, °F	630
Maximum fuel-centerline temperature, °F	4500
Average fuel-surface heat flux, Btu/(hr)(sq ft)	89,000
Maximum fuel-surface heat flux, Btu/(hr)(sq ft)	312,000
Total power, peak/average	3.5
Estimated fuel lifetime, Mwd per metric ton of U	10,000
Estimated fuel lifetime, years	4.06
Plant factor	0.60*

*Specified by AEC.

Table I-5 DESIGN CHARACTERISTICS[7] OF ORGANIC-MODERATED REACTOR

Power Output	
Net electrical kilowatts	20,000
Gross electrical kilowatts	22,000
Gross thermal kilowatts	73,000
Steam Generators	
Steam pressure at turbine throttle, psia	800
Steam temperature, °F	580
Steam rate, lb/hr	272,000
Feed-water temperature at full load, °F	375
Coolant flow to each steam generator, lb/hr	3.37×10^6
Coolant temperature entering superheater, °F	625
Coolant temperature drop in superheater, °F	5
Coolant temperature entering boiler, °F	620
Coolant temperature leaving boiler, °F	552
Feed-water temperature entering boiler, °F	375
Steam pressure at superheater outlet, psig	800
Pressure drop through superheater, psi	20
Steam temperature leaving superheater, °F	590
Steam flow leaving each generator, lb/hr	136,000
Reactor Cooling	
Coolant	Santowax-R
Coolant inlet temperature, °F	552
Coolant outlet temperature, °F	625
No. of primary loops	2
Total coolant flow rate, gal/min	14,700 (6.75×10^6 lb/hr)
Average coolant temperature rise in core (full power), °F	73
Coolant-system design pressure, psig	300
Nominal operating pressure (inlet to core tank), psia	120
Coolant velocity through fuel element (maximum), ft/sec	15.3
Average heat flux in the core, Btu/(hr)(sq ft)	32,400
Maximum heat flux in the core, Btu/(hr)(sq ft)	94,000
Pressure drop through the core, psi	49
Heat-transfer coefficient, Btu/(hr)(sq ft)(°F)	1298
Main coolant pump capacity (one in each loop), gal/min	7350 (each)
Fuel-element Characteristics	
Type	Concentric tubes
Tubes per element	2

Table I-5 (Continued)

Cladding material and thickness, in.	Finned aluminum, 0.035
Element outside diameter, in.	5.25
Element inside diameter, in.	3.10
Thickness of fuel tube, in.	0.208
Fuel-can material and thickness, in.	Stainless steel, 0.030
Fuel alloy	3.5 wt.% Mo-U alloy +0.5 wt.% Si
Active length of fuel element, in.	78
Total length of fuel element, in.	104
Reactor Characteristics	
U^{235} loading, kg	200
U^{235} enrichment, %	1.7
No. of fuel elements	97
Control elements	13 B_4C tubular elements, driven by magnetic jacks

aluminum powder metal (APM) nor subcooled nucleate boiling is utilized. The reference does consider the use of APM in connection with a discussion of the future potential of the reactor type; it is estimated that the reference design modified by inclusion of APM fuel elements might be capable of producing 35.8 Mw(e) gross. It is also stated that radiation-damage-inhibiting additives placed in the organic coolant would probably increase the stability of the organic to break down under irradiation and decrease organic replacement cost; although no details are given in reference 7, the subject has been discussed elsewhere.[8]

The basic fuel-element constituent of the reference design is the uranium alloy tube clad with aluminum; the element is quite similar to the Piqua reactor fuel element. The aluminum fins are twisted slightly to form a spiral along the longitudinal axis of the element. Two fuel cylinders compose the element, and the fins on the annulus side of the cylinders twist in opposite directions. This aids in heat transfer and also in maintaining the dimensions of the annulus.

Pressurized-water Reactor (PWR). Reference 9 surveys PWR power plants in the 10- to 30-Mw(e) size. Five conceptual designs are presented:

1. 12.5 Mw(e), 4 per cent enriched.
2. 11.7 Mw(e), two region (based on Mol BR-3).
3. 25 Mw(e), 3.7 per cent enriched (up-rated version of item 1).
4. 25 Mw(e), highly enriched.
5. 25 Mw(e), 4 per cent enriched with oil-fired superheat.

Plants 1 and 2 were treated in some detail, and unit power-generating costs for the plants were estimated. The estimated cost for the 25-Mw(e) plant with fossil-fuel superheat was less than that for the other designs, and it will be discussed in greater detail. A summary of the design data for the plant is contained in Table I-6.

Power-cost Estimates. Although cost data were presented in all the conceptual design reports, the adjusted data presented in reference 4 are of the most interest, since there were some differences in cost accounting techniques and inconsistencies in design procedures among the original studies. The adjusted comparative electrical generating costs are given in Table I-7. For comparison it was concluded that a conventional plant operating under the same set of ground rules would exhibit a total generating cost of about 10.9 mills/kw-hr. Of this, 3.1 mills/kw-hr represents fixed charges, and 7.8 mills/kw-hr represents fuel cost, operation, and maintenance. The figure 3.1 was computed, however, on the basis of a fixed charge rate of 5.1 per cent compared to the 7.0 per cent used for the nuclear plants. The reason for the lower figure was reduced insurance costs (a factor of 4) and a longer amortization period (35 years as compared to 25 years for the nuclear plant). Although the lower insurance cost may be real, the longer amortization period seems somewhat artificial.

Referring to the generating costs given in Table I-7, the following general comments are given in reference 4:

> Cost Comparison: None of the three plants appears to possess a clear-cut economic advantage over the others for a comparable state of technology at the present stage of their development. Although the BWR plant is estimated at the lowest total generating cost, the difference between its estimated total and that for the PWR is too small (about 1 mill) to be significant when the uncertainties of the estimates are taken into account. Furthermore, the advancement of technology which must be successfully achieved is most extreme in the case of the BWR plant. It is therefore apparent that the cost evaluations, although useful, do not in themselves

Table I-6 DESIGN CHARACTERISTICS[7] OF PRESSURIZED-WATER REACTOR WITH OIL-FIRED SUPERHEAT

[Plant Rating: 25 Mw(e)]

General	
Power output (heat), Mw	57.1
Pressure, psia	2200
Mean temperature, °F	502
Coolant, moderator and reflector	H_2O
Fuel element	Rectangular bundle of fuel pins
Fuel pins	UO_2 pellets in S.S. tube
Control rods	13 movable fuel elements with absorbing followers, bottom driven
Structural	
Core diameter (equivalent), in.	48
Total no. of fuel elements	61
No. of stationary elements	48
No. of control-rod elements	13
Fuel-element array	9 × 9
Fuel-element cell dimensions, in.	5.355 × 5.355
Fuel-element length (active), in.	48
No. of pins per stationary element	81
No. of pins per control-rod element	64
Total no. of pins	4720
Pin outside diameter, in.	0.400
UO_2 pellet diameter, in.	0.356
Pin wall thickness, in.	0.020
Pin center-to-center spacing, in.	0.595
Heat Transfer and Fluid Flow	
Thermal and hydraulic flow rate, gal/min	12,000
Inlet temperature, °F	484.5
Outlet temperature, °F	519
Temperature rise, °F	34.5
Total heat-transfer area, sq ft	1978
Maximum heat flux, Btu/(hr)(sq ft)	3×10^5
No. of passes	2
First pass, no. of elements	Center, 29
Second pass, no. of elements	Outer, 32
Maximum surface temperature, °F	625
Maximum water velocity, ft/sec	6.9
Nuclear	
Initial uranium loading, kg	3335
Enrichment, %	4.0
Initial U^{235} loading, kg	133.4
Initial UO_2, kg	3806
Temperature coefficient, K/°F	-2×10^{-4}
Core exposure, Mwd per metric ton of UO_2	10,000
Lifetime, Mw-years	104.9
U^{235} burn-up, %	39
Total Pu present at end of life (approximate), kg	8.4
Total U^{235} present at end of life, kg	80.9
U^{235} enrichment at end of life, %	2.47

provide a basis for discrimination between the various concepts.

General Comments. Within this framework, the following general comments on each concept appear appropriate:

1. The stated potential advantages of the direct-cycle BWR, i.e., the attainment of lowest over-all generating costs with high thermal efficiency, are also indicated by this estimate. The assurance of obtaining these advantages is somewhat obscured by the lack of sufficient experience with fuel-element defects or failures, other than pinhole defects in the element. Direct-cycle fossil-fuel superheat introduces both a potential corrosion problem with oxygenated steam (because of the high thermal stresses in the superheater) and an operating concern (because of a desire to avoid steam leaks in a direct-cycle system). These problems are in addition to those arising solely from the coupling of two independent heat sources which

Table I-7 COMPARATIVE ELECTRICAL GENERATING COSTS[4]

[Station Rating: 23.5 Mw(e) (Net); Load Factor: 60 Per Cent; Annual Fixed Charge Rate: 7 Per Cent; Oil Cost: $0.50 per 10^6 Btu]

Cost element	BWR	OMR	PWR
Capital cost, dollars	10,931,000	11,272,000	10,448,000
Fixed charges, mills/kw-hr	6.2	6.4	5.9
U^{235} value, initial dollars per kg of U	176.80	182.75	535.50
U^{235} value, final dollars per kg of U	(33.50)	(112.80)	(292.38)
Pu value, final dollars per kg of U	(67.33)	(22.44)	(26.18)
Net burn-up, dollars per kg of U	75.97	47.51	216.94
Fabrication, dollars per kg of U	280.00	70.00	140.00
Reprocessing, dollars per kg of U	64.47	32.86	58.45
Use charge, dollars per kg of U	20.18	25.11	67.75
U^{235} loss, dollars per kg of U	0.34	1.13	2.92
Shipping, dollars per kg of U	10.00	10.00	10.00
Total nuclear fuel, dollars per kg of U	450.96	186.61	496.06
Total nuclear fuel, mills/kw-hr	3.4	4.2	4.4
Conventional fuel, mills/kw-hr	1.8	1.8	2.1
Organic make-up, mills/kw-hr		0.5	
Operation and maintenance, mills/kw-hr	2.0	2.0	2.0
Total, mills/kw-hr	13.4	14.9	14.4

exist even for the use of a fossil-fuel superheater in a secondary circuit.

Nevertheless, inasmuch as high-pressure BWR construction costs are considered to be uncertain at this stage of development, the successful low-cost construction and operation of the proposed plant would be a major step forward in the development of BWR technology as well as nuclear power in general.

2. During the period when nuclear superheat technology and associated costs are still speculative, the excellent steam conditions from the OMR associated with a low-pressure primary loop and the possibility of minimum containment compared to other types make it an extremely attractive concept. However, the lack of construction experience of a true prototype places it behind the other two concepts with regard to the breadth of technological assurance desirable for this particular program. In particular, the problems associated with metallic fuel-element characteristics and life, in association with the desire to push the present coolant to its maximum limit of reliable operability, need to be demonstrated through construction and operation of a prototype such as Piqua.

Actual first-core operation and the research and development effort planned for the next year or so would place the OMR concept into the SSNPP program on a much firmer basis under more appropriate timing as a second-generation plant. The Piqua experience should also allow for resolution of the proper magnitude of second-generation construction costs. Intuitively, the OMR capital estimates appear somewhat high at this time, perhaps as the result of high safety factors and design contingencies presently necessary because of lack of sufficient construction and operating information.

3. The PWR concept has the most actual construction experience behind it to support quotations on equipment and the general reliability of its technology. Although some of this experience is not specifically applicable to the present plant concept, PWR technology has advanced to the point where construction-cost reductions, through relaxation of unduly stringent component specifications and elaborate designs, may now be indicated by design, construction, and maintenance experience.

The PWR concept offers excellent load-following characteristics and a reliability and simplicity of total plant operation presently unmatched by any other competing reactor system. This advantage is obtained at the expense of a primary-secondary circuit design and attendant thermodynamic loss which may result in less favorable economics in the long term. Nevertheless, the construction of such a plant today, at the estimated capital and operating costs, would afford a major competitive target for second-generation plants at a time when such a target is badly needed.

From the above, we conclude that major consideration for the first plant must be limited to the BWR and PWR concepts because of the lack of sufficient prototype experience with the OMR concept in general and the minimal program advantages of duplicating Piqua with an SSNPP on very nearly the same time scale.

The final recommendation of the evaluation is that the first project under the SSNPP program be the PWR plant using slightly enriched fuel and an oil-fired superheater. The reasoning behind this recommendation is quoted as follows:

BASIS OF SELECTION OF PWR FOSSIL-FUEL SUPERHEATER PLANT

1. Reliability of Operation: Since this program has not been described as a reactor experiment, it appears that the publicly owned generating organization would desire some assurance that the plant will be operable upon demand without a long period of experimental start-up and, during its life, without the requirements of excessive downtime for equipment replacement, modification, or decontamination. This is particularly important in view of the limited technical background of the actual plant operators who would normally be available to a small publicly owned organization for operation of either a nuclear or conventional small-sized power station. Continuity of operation is also an important AEC objective, since the experimental nature of a project or the inability to obtain long periods of routine operation would preclude or delay the compilation of reliable cost data which is vitally needed in the power reactor development program. We do not wish to suggest that the BWR, for example, would not operate satisfactorily under such conditions. The question is rather one of the degree of assurance possible before the fact.

Even the PWR plant proposed by Alco has certain novel features which could introduce unanticipated problems in the successful low-cost construction and operation of the plant. The risks, however, are felt to be minimized greatly while at the same time affording measurable advancement of the art through the choice of the PWR with fossil-fuel superheat for construction at this time.

2. Investment in Conventional Turbogenerating Facilities Adequately Protected: Since our concept of this program is based on the participation of a public generating organization through their provision of the conventional turbogenerating facilities, it is reasonable to assume that they will desire to protect this investment to the maximum extent possible. With the reactor system supplying superheated, rather than saturated, steam, the future utility of the turbine-electric system, should it be

separated from the nuclear system, is measurably increased. Perhaps even more important, under this general heading, is the question of superheater and turbine contamination in a direct-cycle system. Despite the excellent experience with EBWR and VBWR and the large-scale commitments to Dresden and SENN, we feel that unreserved acceptance of a direct-cycle system must follow a period during which the problem of operation of such systems at high power levels and power densities is a matter of reasonable concern. This problem is intimately connected with that of fuel-element integrity. One of the major virtues of a closed-cycle system is the ability to continue operation with more than pinhole penetrations of the fuel-element cladding. Once again, with limited experience with direct-cycle plants, the possibility of expensive turbine contamination on the one hand or attendant shutdowns on the other may be over emphasized but cannot be ignored, particularly by a small generating organization such as a municipality.

3. Potential for Maximum Fuel Utilization: The achievement of long average fuel lifetimes is hampered near the end of the core life by the statistical failure of elements and the high maximum exposure in certain parts of the core arising from the peak to average relationship. Thus the indirect-cycle nature of the PWR favors greater fuel-element burn-up for the same fuel element than does the direct-cycle BWR, due to confinement of radioactive containment in the PWR primary cycle. A corresponding advantage in relative fuel cost should therefore exist for the PWR resulting from maximum utilization of the fuel value before discharge.

4. Lack of Duplication of Other Phases of AEC Program: Just as the present status of the art for the OMR concept would result in a slight modification of Piqua for the SSNPP, so a less advanced approach to the BWR than that proposed by GNEC would result in the construction of a near duplicate of other AEC projects such as Elk River. It is significant, however, that no small slightly enriched PWR plant is being built today as a part of AEC's civilian program while the present evaluation indicates that this concept is in fact competitive with other nuclear systems. It is hard to conceive of a stronger reason for initiating immediate construction of a PWR directed specifically towards small power station usage.

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Previous issues of *Power Reactor Technology* have considered the use of nuclear energy in the propulsion of merchant ships.[1,2] References 3 and 4 are reports on the application of the Loeffler type boiler system and the organic-moderated reactor (OMR), respectively, to nuclear propulsion.

Loeffler Type Boiler System

In the Loeffler type boiler system,[3] superheated steam is utilized to produce saturated steam by mixing the two streams in a Loeffler boiler. The saturated steam is then introduced into the reactor, by means of blowers, wherein superheat is added. A portion of the superheated steam is used in the boiler, and a portion is sent to the propulsion turbines; the system is, of course, direct cycle. The reactor moderator is the feed water to the Loeffler boiler from the last-stage feed-water heater. The system is designed to produce steam at about 850 psi and 850°F, typical of present-day merchant-ship power plants. No physics calculations are included in reference 3, but subsequent investigations indicate that the design is approximately correct.

The plant performance and design data are given in Table II-1. The moderator of the reactor is water, and a beryllium oxide–water reflector is employed. The basic unit in the fuel-element design is the fuel rod or capsule, composed of a 304 stainless-steel tube filled with UO_2 pellets or slugs. Saturated steam is superheated in one pass through the fuel bundle, which is composed of seven rods. Besides being used to drive the main propulsion turbine, the superheated steam is also used to drive the ship's service turbogenerator, the steam blower turbines, and a heat exchanger to produce auxiliary uncontaminated ship's service steam.

A fairly detailed section on hazards is included in reference 3. Among the questions considered are those related to possible contamination of the direct-cycle steam system. Continued operation with defected fuel elements could contaminate the turbine, and associated steam piping, to an extent which would require a considerable amount of shielding. This difficulty could be circumvented by employing an intermediate heat exchanger, at a sacrifice of system efficiency and simplicity. The other accidents are treated in somewhat lesser detail, and the author's conclusions serve to illustrate the problems:[3]

> The direct-cycle steam-cooled reactor plant is believed to be better suited for central-station applications than for marine propulsion service. The more or less "standard" merchant-ship steam conditions are limited to approximately 850 psi at 850°F, and the performance of the reactor system described in this report is not particularly impressive at these relatively low pressures and temperatures. The consequences of a fuel-element rupture would be much more difficult to deal with aboard ship than in a central-station generating plant. Many essential auxiliary equipment items must, of necessity, be located in close association with the main steam system, and these would almost certainly be inaccessible for operation or repairs if the main steam system should become highly contaminated. These problems would be much less objectionable in a land-based power station, in which a power-plant outage with gross system contamination would never have the disastrous consequences that it might have at sea during a storm, for example. In the event of a reactor accident, the removal of fission-product decay heat from the fuel elements would be more difficult to accomplish aboard ship, partly because of the limited headroom available for the design of a good natural-convection system, and partly because aboard ship it would be difficult to apply emergency measures to effect a makeshift emergency forced-circulation coolant flow.
>
> Further developmental work would be required on at least the following items before technical feasibility of the plant could be established: (1) insulation for the coolant tubes, (2) fuel-element supporting structure, and (3) thermally stable fuel elements. The fact that the steam must be used at high pressure for it to be an attractive reactor coolant tends to restrict the steam-cooled reactor to a relatively small pressure vessel. The development of a satisfactory pressure-tube design would be a solution to

Table II-1 PLANT PERFORMANCE AND DESIGN DATA[3]

General

Reactor thermal power, Mw	73
Shaft power, SHP	20,000
Reactor steam flow, lb/hr	964,000
Propulsion-turbine steam flow, lb/hr	150,240
Propulsion-turbine heat rate, Btu/(hp)(hr)	7360
Moderator	Water
Core height, ft	10
Core diameter (active), in.	42
No. of fuel channels	110
Lattice pitch (triangular), in.	3.55
Pressure-tube material	M-388 aluminum alloy
Pressure-tube OD, in.	3.425
Pressure-tube wall thickness, in.	0.0625
Fuel-channel ID, in.	2.770
Fuel-channel insulation	Aluminum oxide
Insulation thickness, in.	0.250
Insulation liner	Type 304 stainless steel
Liner thickness, in.	0.010
Side reflector	BeO (Al clad) + H_2O
BeO reflector thickness, in.	5.5
H_2O reflector thickness, in.	7.0

Fuel Element

Fuel composition	Partially enriched UO_2
UO_2-slug OD, in.	0.71
UO_2-slug ID, in.	0.32
UO_2-slug length, in.	0.50
UO_2 density, g/cm^3	10.4
No. of slugs per capsule	75
Capsule material	Type 304 stainless steel
Capsule OD, in.	0.75
Capsule wall thickness	0.020
Capsule length, in.	40
No. of capsules per element	7
No. of elements per channel	3
Element external surface, sq ft per foot of element	1.375

Reactor Pressure Vessel

Shape	Cylindrical
Inside diameter, in.	68
Over-all height, ft	18
Wall thickness, in.	2
Material	Carbon steel (stainless-steel-clad inside)
Stainless-steel cladding thickness, in.	0.25
Design pressure, psia	900
Design stress, psi	15,000

Core Flow and Heat Transfer

Coolant	Steam
Reactor inlet pressure, psia	880
Core pressure drop, psi	16.1
Steam velocity (average), ft/sec	131
Channel hydraulic diameter, in.	0.472
Maximum design surface temperature, °F	1200
Steam inlet temperature, °F	525
Steam exit temperature, °F	860
Axial power, maximum/average	1.43
Radial power, maximum/average	1.5
Over-all power, maximum/average	2.15
Maximum heat flux, Btu/(hr)(sq ft)	340,000
Average heat flux, Btu/(hr)(sq ft)	158,000

Steam Blowers

Type	Single-stage centrifugal
No.	3 (2 operating and 1 spare)
Inlet pressure, psia	850
Pressure rise, psi	30
Capacity, cu ft/min	4530
Pumping power per blower, hp	750
Speed, rpm	3600
Drive	Steam turbine

this problem, however. Since steam is a chemically reactive coolant, problems of compatibility of coolant, moderator, fuel, and structural materials could be considerably greater than in the case of the inert-gas-cooled reactor.

Organic-moderated Ship Reactor

Reference 4 reports on the application of the OMR to marine propulsion. Reference 2 discussed the pressurized-water and boiling-water reactors as applied to marine propulsion, and the current discussion can be considered as a supplement to the earlier one.

The study was based on the use of a type T-7 tanker for the nuclear application. The T-7 tanker has a total dead weight of 60,840 tons and requires 27,250 SHP for normal operation and 30,000 SHP maximum power. The trip studied for economic potential consisted of a 21,350-mile round-trip voyage from Philadelphia to Kuwait.

The propulsion system was designed to meet the following operating conditions:

Permanent list	15°
Temporary roll	45°
Pitch	10°
Shock loading:	
Transverse	1.1 g
Perpendicular	2.0 g
Longitudinal	0.2 g

The basic ship's power is derived from one OMR, which furnishes propulsion steam, steam for generation of ship's electricity, and hotel heating. The OMR vessel, the heat exchangers, and the auxiliary reactor equipment are located within a type of containment room within the ship. The containment structure is formed by a cofferdam surrounded with structural-steel stringers to act as a collision mat. The top and bottom of the containment structure are decks; the reactor compartment is designed to contain an internal pressure of about 15 psig. This corresponds to the maximum expected pressure caused by a major leak in the reactor system.

The primary shield surrounds the pressure vessel and attenuates the dose from the reactor to that from coolant piped exterior to the pressure vessel. The primary shield is borated water and lead. Surrounding the reactor compartment is the secondary shield. The secondary shield serves to reduce the dose rate from the reactor primary shield and coolant to a level of 1.5 mrem/hr. The secondary shield is composed of steel shells (actually part of the cofferdam and decks) and oil or water.

The basic constituent in the fuel element is the fuel rod or pin. The fuel is enriched UO_2, formed into cylindrical pellets 0.6 in. long by 0.33 in. in diameter. These pellets are stacked vertically to form an active core height of 60 in. and are contained in finned tubing made of aluminum powder metal (APM), Alcoa alloy M-257. One hundred rods are enclosed in a stainless-steel box to form a fuel element $5\frac{1}{8}$ in. on a side by about $6\frac{1}{2}$ ft in length. Organic coolant enters at the top of the thermal shield and flows around the shields to a plenum at the bottom of the reactor. After passing through the reactor in upflow, the heated coolant is pumped through a superheater and the steam boiler before returning to the core; steam is generated at a rate of 150,000 lb/hr and is 450 psig and 650°F at the superheater outlet. The maximum cladding temperature is set at 825°F with the maximum fuel-pellet centerline temperature set at 4000°F. Control is achieved by 21 cruciform control elements for shim, safety, and radial flux shaping. The poison material is europium oxide clad with stainless steel, and the rods are followed by cruciform-shaped APM followers. The rods are bottom driven but may be removed and replaced, along with the fuel elements, from the top of the reactor. The upper-grid and control-rod-guide assembly locks the fuel elements in place in the core and conducts tubing from groups of fuel elements to the failed-fuel-element detection and location system. Other details of the reactor can be found in Tables II-2 and II-3.

Table II-2 THERMAL AND FLOW DATA FOR CORE, ORGANIC-MODERATED SHIP REACTOR[4]

Coolant temperature, °F:	
Inlet	620
Mixed mean outlet	675
Maximum local outlet (with hot-channel factors)	703*
Saturation at center of core	775
Fuel-rod temperature, °F:	
Maximum cladding surface	825
Maximum UO_2 surface	1450
Maximum UO_2	4000
Thermal power of reactor, Mw	93
Power transferred across fuel-rod surface	86.5
Power generated in coolant, thermal shields, etc.	6.5
Heat fluxes, Btu/(hr)(sq ft):	
Burn-out	250,000
Maximum	125,000
Average	28,700
Heat-generation rate in fuel pins, kw/ft:	
Maximum	8.56
Average	1.97
Surface area of fuel rod, sq ft/ft	0.0243
Effectiveness of surface area	0.96
Pressure at center of core, psia	34
Pressure drops, psi:	
Core	12
Reactor tank, exclusive of core	3
Total pressure drop from inlet to outlet nozzles	15
Flow rate, lb/hr:	
Through fuel elements	10.96×10^6
Leakage around fuel elements	0.34×10^6
Total flow rate	11.3×10^6
Velocities in fuel elements, ft/sec:	
Maximum	10.0
Average	6.9
Minimum	5.0

*This figure assumes no coolant mixing in the fuel elements. Since there will be mixing, the actual maximum local coolant outlet temperature will be somewhat lower.

The reactor design allows subcooled boiling in those fuel elements where the heat flux is sufficiently high. The subcooled boiling (in the hotter channels) and the APM cladding are both design features that were employed in the large-power-reactor studies reported in reference 5 and discussed in reference 6.

Reference 4 recognizes the need for research and development programs in these, and other, areas and recommends the following research and development programs:

> The organic-moderated and -cooled reactor concept as proposed in this report represents a signifi-

Table II-3 DESIGN FEATURES OF REACTOR VESSEL AND STRUCTURE, ORGANIC-MODERATED SHIP REACTOR[4]

Reactor core tank:	
Over-all height, including heads	21 ft 11 in.
Inside diameter	9 ft 3 in.
Design pressure, psig	350
Design temperature, °F	750
Main coolant nozzles, 18 in. in diameter	2
Instrumentation nozzle, 14 in. in diameter	1
Drain nozzle, 4 in. in diameter	1
Instrumentation nozzles, 16 in. in diameter	2
Wall thickness, in.	$1\frac{1}{2}$
Weight of vessel, lb	72,000
Outer thermal shield:	
Inside diameter, in.	97
Wall thickness, in.	6
Height, in.	96
Weight, lb	70,400
Inner thermal shield:	
Inside diameter, in.	73
Height, in.	71
Wall thickness, in.	1.5
Weight, lb	6960
Core support structure:	
Inside diameter, in.	78 and 93
Height, in.	162
Wall thickness, in.	1.5
Weight, lb	17,720
Lower grid plate:	
Diameter, in.	77.5
Weight, lb	3400
Upper grid and control-rod guide:	
Diameter at bottom support, in.	76
Height, in.	56
Weight, lb	5000
Fuel elements:	
No. of subassemblies per element	4
Length of subassemblies, in.	67
No. of fuel rods per subassembly	25
Fuel-rod lattice spacing, in.	Variable
Active length of fuel rods, in.	60
Over-all length of fuel element, in.	78.25
Coolant flow area per fuel element, sq in.	13.6
Outside dimension of fuel-element box, in.	5.125
Wall thickness of S.S. fuel-element box, in.	0.035
Total weight of fuel per element, kg	89.1
Total weight of uranium per element, kg	78.6
Total weight of fuel element, lb	279
Control rods:	
Poison material	Eu_2O_3
Cladding material	Stainless steel
Length of poison section, in.	54
Total length of blade section, in.	66
Control-rod shape	Cruciform
Blade length, in.	4.5
Weight of control rod and follower, lb	107
Rod drive type	Rack and pinion
Total weight of rods and drives, lb	307
Reactor core:	
Total no. of lattice positions	96
Average lattice spacing, in.	5.75
Total no. of fuel elements	88
No. of dummy elements	7
No. of neutron-source elements	1
No. of control rods	21
Equivalent core diameter, in.	60
Equivalent core height, in.	60
Equivalent reflector thickness, in.	6.5
Total weight of fuel in the core, kg	7770
Total weight of uranium in the core, kg	6850
Total weight of fuel core loading, lb	24,500

cant advance over other OMR designs. As such, several well-defined basic areas require an extension of present development effort. These areas are: (1) a complete experimental investigation of nucleate-boiling heat transfer with organics, including fouling and burn-out studies; (2) fuel-element materials and design studies, irradiation and postirradiation evaluation of materials, and the investigation and development of fabrication techniques, specifically for APM and similar fuel-element cladding materials; (3) the investigation of the physics of a power flattened core; and (4) determination of pressure build-up characteristics during the suppression of organic fires in contained areas. Component development work deemed necessary includes prototype control-rod testing, testing of the failed-fuel-element detecting system, testing of the incore power mapping system, and development of absolute seals for organic coolants.

Coolant degradation represents a penalty in at least two respects: the cost of replacement and the necessity for disposal or storage of radioactive residue from coolant processing. The waste-disposal system presents novel requirements due to the maritime environment. The end product of the organic purification system is liquid organic waste in the form of high-boiler compounds (HBC); aqueous waste is generated in the coolant degasification and purification systems and the off-gas scrubber; gaseous wastes are formed from the coolant decomposition and may also arise from ruptured fuel elements. Provision has been made in the design to store all liquid wastes generated on a round trip in tanks. HBC can be incinerated at sea and discharged if the activity is sufficiently low. The waste-gas handling system vents gases to the atmosphere so long as their activity is sufficiently low. Upon rupture of a fuel element, however, volatile fission products find their way into the coolant degasification system and hence

into the gaseous wastes. When the gaseous waste-stream activity reaches some limit, the stream is diverted into a waste-gas processing system to separate xenon and krypton from the stream before discharge. A shielded area is also provided for small amounts of highly active liquid and solid wastes.

The cargo-capacity and capital-cost summary for the OMCR tanker is given in Table II-4.

Table II-4 WEIGHTS, CAPACITIES, AND CAPITAL COSTS FOR THE OMCR TANKER[4]

[Weights Are in Long Tons; Costs Are in Thousands of Dollars (Except Capital Cost per Ton of Cargo) Escalated to About 1963]

Item	Cost	Weight
Displacement		77,100
Dead weight		60,840
Cargo capacity		60,310
Total cargo per year		386,000
Conventional tanker cost	$19,200	
Boiler and propulsion machinery	5,083	
Conventional tanker steel and outfit	$14,117	
Reactor and propulsion machinery, including engineering design, barge equipment, and supplies (not including initial core loading)	$14,182	
Secondary shield and containment	203	
OMCR tanker capital cost	$28,502	
Capital cost per ton of cargo	$473	

Table II-5 OPERATING COST SUMMARY[4]

	OMCR tanker	Conventional tanker
Crews' wages	$ 500,000	$ 435,000
Subsistence	47,000	40,800
Stores and supplies	134,000	30,000
Maintenance, repair, and refueling	215,000	180,000
Insurance (not including nuclear liability	360,000	230,000
Port expenses	45,000	45,000
Overhead and miscellaneous	53,500	53,500
Canal tolls	7,700	7,700
Interest and depreciation	2,285,000	1,540,000
Total cost, excluding fuel	$3,647,200	$2,562,000
Fuel cost	725,000	876,000
Total annual operating cost	$4,372,200	$3,438,000
Long tons of cargo per year	386,000	360,000
Cost per ton of cargo	$11.30	$9.55

The fuel costs for the first core were based on an average burn-up of 10,000 Mwd/ton. The estimated steady-state fuel cost was 3.57 mills/(SHP)(hr) for the reference-design conditions. It was estimated that some possible improvement in the steam cycle, plus a 40 per cent reduction in fuel-fabrication cost, plus an extension of fuel life to 20,000 Mwd per metric ton of uranium might decrease the fuel cost to 2.73 mills/(SHP)(hr). The estimated yearly operating costs for the OMCR tanker and a conventional vessel are given in Table II-5 for a nuclear fuel cost of 3.57 mills/(SHP)(hr) and a fuel-oil cost of $2.70 per barrel. Depreciation has been taken at 8.02 per cent. It was estimated that the second OMCR tanker might be approximately $5,000,000 cheaper than the first tanker. This cost reduction is estimated by assuming:

1. The cost of engineering and design is reduced by 67 per cent.
2. Charge for manuals is eliminated.
3. Barge costs are eliminated.
4. Service reserve is reduced by 80 per cent.
5. Equipment and installation costs of the steel, outfit, and machinery are reduced 10 per cent.

The assumption of a 40 per cent reduction in fuel-element fabrication cost reduces the fuel costs from 3.57 to 3.07 mills/(SHP)(hr). This reduction, combined with the lower capital cost, reduces the estimated cost of hauling cargo in the second OMCR tanker to $10 per ton. This cost is 5.8 per cent greater than that for the conventional tanker which consumes fuel oil worth $2.70 per barrel; an increase in fuel-oil cost to $3.20 per barrel would make the estimated second OMCR tanker competitive.

References

1. *Power Reactor Technology*, 2(2): (March 1959).
2. *Power Reactor Technology*, 2(3): (June 1959).
3. H. L. Falkenberry, R. S. Holcomb, and G. Samuels, A Preliminary Study of a Direct-cycle Steam-cooled Reactor for Merchant Ship Propulsion, ORNL-2759, Sept. 23, 1959. (Unclassified AEC report.)
4. Maritime Organic Moderated and Cooled Reactor, NAA-SR-3859, May 27, 1959. (Unclassified AEC report.)
5. Bechtel Corp. and Atomics International, Organic-cooled Reactor Study, TID-8501, April 1959. Part 1, Summary of Study (previously available as AEC Report BCPI-1, Vol. 1); Part 2, 300 Mw Power

Plant Conceptual Design (previously available as AEC Report BCPI-1, Vol. 2); Part 3, Reactor Concept Evaluation (previously available as AEC Report BCPI-1, Vol. 3); Part 4, 75 Mw Power Plant Conceptual Design (previously available as AEC Report BCPI-1, Vol. 4); Part 5, 300 Mw Coal-fired Power Plant Comparison Study (previously available as AEC Report BCPI-1). (Unclassified AEC report.)

6. *Power Reactor Technology*, 2(4): (September 1959).

COSTS AND SUPPLIES OF ENRICHED URANIUM

In his address to the Atomic Industrial Forum on Nov. 4, 1959, Chairman McCone of the Atomic Energy Commission invited the Forum to form a committee to make an independent evaluation of the present uranium-enrichment costs and the present capabilities of the enrichment plants for supplying enriched fuel to commercial power reactors. He indicated that the committee would be given access to the classified information necessary for such an evaluation. The committee was appointed, and it consisted of Kenneth D. Nichols, Consultant and former General Manager of the AEC, William Webster, President of the New England Electric System and the Yankee Atomic Electric Company, and Percival Brundage, a retired partner in the accounting firm of Price, Waterhouse and Company and former Director of the Bureau of the Budget. The committee has made its report in a letter to Forum President Francis K. McCune.[1] The main conclusions of the committee are summarized as follows:

1. The published AEC prices for U^{235} (in varying degrees of enrichment) do not include any government subsidy. It is the opinion of the committee that the future price of U^{235} in terms of 1959 dollars should not increase, and a decrease might be warranted.

2. At the present level of interest rates, the use charge to the industry of 4 per cent per annum does contain an element of subsidy.

3. If weapon production requirements should terminate as a result of disarmament agreements, the increased cost of partial operation of existing plants sufficient to meet the anticipated power and other nonweapon uses would not require any increase over the present published prices.

4. Taking into account anticipated U^{235} requirements (both military and commercial), ore reserves, and U. S. refining capacity and U^{235} separation capacity, the committee concludes that there can be (enriched) U^{235} available to meet both the inventory requirements and the continuing burn-up requirements for a commercial atomic power capacity of at least 40 million kilowatts.*

5. No new major construction of separation capacity will be needed until the nuclear power demand reaches that indicated above, probably not for 15 years at the earliest. When additional separation capacity is needed, a separation plant and the conventional electrical capacity to supply the plant could be built to supply low-enrichment (up to 3 per cent) uranium fuel for 20 to 30 million kilowatts of nuclear power plants for a capital cost of about 500 million dollars (about $20 per kilowatt of nuclear electrical capacity). Such a separation plant would produce U^{235} at costs within the limits of existing AEC published prices. The cost studies for this plant assumed fixed charges of 14 per cent on investment capital, feed costs of $8 per pound of U_3O_8, electric power costs of 4 mills/kw-hr to run the separation plant, and operating costs in line with those of existing separation plants. If electric power for this separation were supplied by nuclear power plants, the U^{235} costs might exceed present published prices by not more than 10 per cent.

It is no great surprise that the prices of U^{235} are not subsidized in the literal sense of being less than the production cost. The major economic advantages that the users of enriched fuel presently enjoy do not result from direct government subsidies, but from the fact that the government carries the capital burden for the enrichment plants and chemical processing plants which are essential to the use of enriched fuels and which could not be supported commercially until the nuclear power industry has grown large. As an accompanying disadvantage, the users must accept the fact that their fuel costs are, to a large extent, determined by government regulation. If the estimates of the Forum committee are correct, perhaps it could

*At 80 per cent load factor, a capacity of 40 million kilowatts would supply about 30 per cent of the present average output of utilities in the United States.

be hoped that the separation of isotopes for reactor use might become a feasible commercial venture by about the time the installed nuclear power capacity reaches 40 million kilowatts, provided that military considerations do not require the government to retain control of all separation capacity. If nuclear fuels were handled in a free commercial market, the use charges would, of course, be replaced by capital charges at appropriate commercial rates (cf. discussion of capital charges for D_2O inventory in the section on Heavy-water Reactors in this issue).

There does not appear to be any good reason why government facilities that are already in existence because of military needs should not be used to reduce the capital outlay necessary for the growth of the nuclear power industry. This use does, of course, exert an artificial force in determining the direction of development of nuclear plants. It is unfortunate that, in the case of the isotope-separation plants, this force tends to direct the development toward rather wasteful exploitation of the nuclear fuel resources. It is not yet clear that the use of relatively high enrichments (about 3 per cent) in "thermal" reactors is necessarily incompatible with good neutron economy, although this may be true because of the relatively high values of capture to fission ratio for fissionable isotopes in the low epithermal energy range. However, the availability of enriched fuel at the low prices which are characteristic of a large-volume production does present the temptation to use enrichment indiscriminately, wherever it may appear to improve the transient economic picture of the low-volume nuclear power industry. Such uses of enrichment generally lead to poor neutron economy and wasteful exploitation of the fuel resources.

Reference

1. *The Forum Memo to Members*, Atomic Industrial Forum, Inc., 6(12): (1959).

A recent conference at Argonne National Laboratory on the Physics of Breeding has focused attention on the subject of breeding and breeder reactors. The conference papers and other papers relating to the subject are discussed briefly in the following.

Significance of Breeding

The generally recognized importance of breeding is that it can extend by a large factor the nuclear fuel supply. It is perhaps more to the point to say that by breeding we can multiply by a large factor the amount of energy obtainable from each ton of uranium extracted from the earth. Such a statement is consistent with the apparent circumstance that the practical limit on the supply of fissionable fuel will not be determined by the actual amount of uranium and thorium in the earth's crust, but rather by the amount which can be extracted economically. The upper limit on the acceptable cost of extraction is, of course, determined by the amount of energy that can be obtained from each ton of uranium or thorium.

The lower cost reserves of uranium and thorium have been discussed in a previous issue of *Power Reactor Technology*[1] and elsewhere.[2] More recently Weinberg[3] and MacPherson[4] have pointed out that much of the uranium and thorium content of the granite crust of the earth should be extractable at sufficiently low cost to constitute low-cost fuel for a breeder cycle. With the addition of this reserve of nuclear fuel, we presently recognize, roughly, three categories of reserves: the high-quality ores of the types which are presently supplying most of the uranium mined, the low-quality ores such as the bituminous shale and phosphate deposits, and the material contained at very low concentration in the general granitic structure of the earth's crust. The estimated reserves in these categories and the estimated cost of mining and extraction are summarized in Table IV-1. No doubt the actual situation is that gradations of ore quality exist between the categories and that, as the nuclear fuel is used up, the cost of extracting fuel from the earth will increase gradually rather than stepwise. The total energy content of the high-quality ore is not very large relative to the present rate of energy use if only the U^{235} is burned. The energy content of all the U^{235} in the earth's crust is, however, large; but to burn only the U^{235} from uranium which is ex-

Table IV-1 ESTIMATES OF FISSION-FUEL RESERVES

Type	Estimated reserve	Electrical energy production if all U^{235} in reserve is burned,* 10^{12} kw-hr†	Electrical energy production if all material in reserve is fissioned,* 10^{12} kw-hr	Estimated cost of mining and extraction
High-quality ores	580,000 tons of U_3O_8 in U. S.‡	22	3,700	\$10 per pound of U_3O_8 or less (reference 2)
Low-quality ores	6×10^6 tons of U_3O_8 in U. S.‡	220	37,000	\$40 to \$50 per pound of U_3O_8 (reference 2)
Earth's crust	16×10^{13} tons of U; 50×10^{13} tons of Th, in total earth's crust[3]	6×10^9	4×10^{12}	\$140 to \$1400 per pound of U (reference 3)‡

*At 30 per cent efficiency.

†10^{12} kw-hr is the order of magnitude of the current annual electrical production by utilities in the United States.

‡Applies to some fraction, perhaps of the order $1/10$, of the total in the earth's crust.

tracted at a cost of some hundreds of dollars per pound would result in a hopelessly high fuel cost.

Roughly, from Table IV-1, there is in the United States (1) a reserve of \$10 per pound U_3O_8 which could be burned rather quickly in a full-scale nuclear power program, (2) a reserve of U_3O_8 which is perhaps 10 times as large and approximately five times as expensive as the \$10 per pound U_3O_8, and (3) an almost unlimited reserve of U_3O_8 which would be available at a cost of about 50 times the present cost. It is convenient to think of these costs in terms of the cost per gram of U^{235} contained in, or extracted from, the natural uranium. The \$10 per pound of U_3O_8 corresponds to approximately \$3.60 per gram of contained U^{235}. If the uranium is used as partially enriched material, and if the mining and extraction cost of the U_3O_8 is all assigned to the enriched U^{235}, then its contribution to the cost of U^{235} will depend upon how much U^{235} is discarded with the low-enrichment stream from the isotope-enrichment plant; but a value of about \$4.60 per gram of U^{235} is perhaps representative for partially enriched fuel. If this contribution of the mining and extraction cost per gram of U^{235} is represented by the symbol c, then the contribution of the mining and extraction cost to the cost of power from the nuclear plant is given by

$$\text{Burn-up cost} = \frac{c}{24\,\epsilon w m}$$

where ϵ is the thermal efficiency and w is the energy, in thermal megawatts, released by the burning of 1 g of U^{235}. The quantity m may be thought of as the multiplication, by breeding or conversion, of the energy obtainable by burning pure U^{235}. It is the ratio of the total energy obtained (from U^{235} and fissionable isotopes produced by conversion) per gram of U^{235} burned to the energy released directly from the burning of 1 g of U^{235} [approximately 0.8 Mw(t)].

In a nonbreeding reactor with complete recycle of the converted fissionable isotope,

$$m = \frac{1}{1 - \bar{R}_c}$$

where R_c is the effective conversion ratio (including losses which occur outside the reactor).

In a breeder reactor system, the equation for m does not hold; if all the fertile material is burned as well as the fissionable isotope, m is equal approximately to the ratio of the total fertile-plus-fissionable isotope to the fissionable isotope, i.e., approximately 140.

In a converter reactor without recycle of converted isotope, m will be somewhat larger than unity, but hardly as large as 2. The power reactors presently being built belong to this class. For the case of $m = 1.0$, $c = \$4.60$ per gram, and $\epsilon = 0.3$, the burn-up cost component resulting from the cost of mining and extracting the U_3O_8 amounts to about 0.8 mill/kw-hr. Since the 0.8 mill/kw-hr represents an appreciable fraction of the acceptable total fuel cost, it might be said that any future large increase in the mining and extraction cost must be at least approximately compensated by a proportionate increase in m (above unity) if the total fuel cost is to remain acceptable. Thus an increase by a factor of 5 in the mining and extraction cost would require that the energy produced per gram of U^{235} burned be increased by roughly a factor of 5; an effective conversion ratio of 0.8 would be required. If fuel mined at a cost 50 times the present cost is to be used, then the energy multiplication (m) must be roughly 50. For this case the effective conversion ratio must be very nearly 1, and when losses of fissionable isotope in chemical processing and other extra-reactor procedures are taken into account, such an energy multiplication will be obtainable only with a breeder reactor.

The fuel cost also contributes to the power cost by way of inventory charges. The contribution of the mining and extraction cost to the inventory cost is given by

$$\text{Inventory cost} = \frac{104\,cei}{pf\epsilon}$$

where e is the enrichment of the fuel, in grams of U^{235} per gram of fuel; i is the fractional annual inventory charge rate; p is the specific power, in thermal megawatts per ton of fuel; and f is the plant factor. For $c = \$4.60$ per gram, $e = 0.03$, $i = 0.04$, $p = 20$ Mw/ton, $f = 0.8$, and $\epsilon = 0.3$, the contribution of mining and extraction cost to the inventory cost is 0.12 mill/kw-hr. This contribution takes into account only the fuel in the reactor; the contribution to the total inventory cost might be higher by a factor of 1.5 to 2.0. So long as the inventory charge rate remains at its present low value, this item of fuel cost is relatively low and some increase

in the mining and extraction cost could be tolerated without an important increase in total fuel cost. However, there is no factor in the inventory cost equation (such as the energy multiplication, m, in the burn-up cost equation) which can be counted on to compensate large changes in c. The inventory cost may represent a much more inflexible limitation on the use of expensive fuel than does the burn-up cost. Although mining and extraction costs of up to as much as five times the present ones might be tolerable from the standpoint of inventory cost, provided the rate of inventory charge does not increase, extraction costs much higher than that would hardly be acceptable, and the U^{235} extracted from the granite of the earth's crust could not be used as a source of inventory unless very large increases in specific power were found to be possible.

Thus, although the supply of fuel suitable for burn-up in breeder reactors may be for practical purposes unlimited, the supply of fuel suitable for use as fissionable-isotope inventory is by no means unlimited. This fissionable-isotope fuel inventory may, however, be sufficiently large to implement a power-production capacity which is large compared to the present capacity in the United States. Recognition of the problem of supplying inventory motivates the efforts toward breeder reactors in preference to nonbreeders of very high conversion ratio and gives some validity to the use of the doubling time as a figure of merit for breeder reactors. Once the supply of natural fissionable isotope which is available at a low enough cost to be usable as inventory has been exhausted, further expansion of the nuclear power capacity must depend for its initial fuel requirements on the net fissionable-isotope production of existing breeders.

The Fast Breeder

The fast breeder reactor can be considered technically feasible in all respects except perhaps the technology of its fuel cycle. There is little question as to the feasibility of operation with uranium fuel elements, although the temperatures and fuel lifetimes attainable may be less than desired; but there has been little experience with plutonium elements, and, of the experience which has been obtained, that with metallic plutonium elements has not been encouraging. Nevertheless, if economic considerations could be completely disregarded, there is little doubt that fast breeders with substantial breeding gains could be built and operated.

The fast breeder does have economic problems. With present technology at least, the plant itself is expensive. To a large extent the expenses result from the fact that in a fast breeder the amount of diluent which can be used with the fissionable isotope (whether the diluent be fertile material or nonfertile material) is relatively small; consequently the power density in the reactor must be quite high if the fuel-inventory costs are to be kept within reasonable bounds. Some of the expenses result rather directly from the high power density; others result from the necessity for using a liquid-metal coolant, which, in turn, results from the requirement for high power density and negligible neutron moderation. It is not entirely clear at present how many of these economic problems are transient ones which can be eliminated through research and development, and how many will persist to influence the cost of fast breeders in the years to come. Some of the basic problems that may have persistent effects on the economics are the following:

1. The necessity for high power density requires very careful design and precise construction of the reactor internals.

2. The high power density magnifies the problems connected with power generation after reactor shutdown. Very careful attention must be given to the problems of shutdown cooling and emergency cooling, and the high power density in the fuel, even after it is removed from the reactor, poses many problems in the handling and processing of the fuel if large fuel inventories external to the reactor are to be avoided.

3. The specific characteristics of sodium generate a number of problem areas: the necessity for heaters on the circulating system to prevent freezing when the reactor is not in operation; the difficulty of producing mechanisms to operate under sodium and the necessity for remote operation of any mechanisms under sodium because of the opacity of the fluid; the extra precautions necessary to prevent contact between sodium and water, particularly between radioactive sodium and water (as exemplified in the rather general use of intermediate heat exchangers); the high level of induced radioactivity in irradiated sodium; the precautions necessary against the possibility of sodium fires.

4. Problems stemming from safety considerations, aggravated by the high power density; the necessity for very careful design and construction of the reactor core to prevent fuel distortions under power which could lead to undesirable power coefficients of reactivity; provisions against the possibility of agglomeration of core material into supercritical masses in case of a core meltdown; the generally more cautious attitude toward fast-reactor hazards because of unique characteristics of the reactor type, either supposed or real.

But paradoxically, it is in the area of fuel costs, where the fast breeder is expected in the future to be preeminent, that the reactor type presently appears to compete least favorably. In some respects it is not reasonable to give much importance to fast-breeder fuel costs until the technology has developed to the point that the reactor is truly operating as a breeder, that is, with plutonium recycle. When fuel-cycle costs based on the use of U^{235} as the fissionable isotope are considered, certain expenses exist which would perhaps not be typical of the plutonium fueled reactor. The breeding gain when U^{235} is used is apt to be small, and if the plutonium produced is valued at $12 per gram, then there may be a net positive burn-up cost which results from the price differential between the plutonium and the rather highly enriched U^{235} even though the amount of plutonium produced is somewhat larger than the amount of U^{235} burned. Further, the inventory cost tends to be relatively high. If the reactor were operated as a pure plutonium breeder, the breeding gain might be high enough to compensate or overcompensate the inventory cost. However, the fuel-fabrication and chemical processing costs also tend to be high, and it may well be expected that, for given processing and fabrication techniques, a shift to the plutonium cycle would only aggravate the situation. One may well ask why these costs are particularly high for the fast reactor when burn-ups of the fuel as high as 2 per cent (about 17,000 Mwd/ton) are visualized. Part of the answer is that both core and blanket elements are involved in the reprocessing and refabricating cycles, and, at least in the current reactor concepts, the blanket elements do not produce a larger amount of energy per ton of material over their lifetimes. The cost of chemical processing is further increased by the fact that the core elements, because they are more highly enriched than those of thermal reactors, must be processed in smaller batches for criticality reasons. Finally, the fuel-fabrication cost is high because the fuel must be highly subdivided to achieve the necessary high power density, and because great care must be taken in the assembly and support of the individual elements to avoid the possibility of power-dependent distortions which might lead to undesirable power coefficients of reactivity.

Two avenues of approach to fuel-cost reductions exist, which may be followed independently or in combination. The first is simply to extend the fuel lifetime by a rather large factor. This approach is part of the motivation for possible changes from the current metallic fuel elements to ceramic elements, such as oxides or carbides, or to cermets. The other approach is to seek simplified methods of fuel processing and fuel-element fabrication. This approach has been followed in connection with the EBR-II plant, which includes facilities for chemical processing by pyrometallurgical techniques and fuel fabrication by casting techniques.

Initially, work on the physics of fast reactors was directed toward determining whether substantial breeding gains were possible and toward the development of theoretical methods for the analysis of fast-neutron chain-reacting systems. There is now no doubt that useful breeding gains are fundamentally possible, and enough experimental measurements have been made to allow the setup and proof of theoretical methods based on the multigroup approach.

A general survey of the physics of fast reactors can be made from references 5 to 7, and further information is contained in references 8 to 11. Experimentally, what is still needed in fast-reactor physics is further detailed information on cross sections and fission properties as functions of neutron energy, although computationally much of the current effort is directed toward investigating reactor designs which may avoid some of the engineering problems.

Some of the Argonne conference papers discussed the status of neutron yield per fission, capture cross sections, and scattering cross sections, in the energy regions important for fast reactors. Terrell[12] reported measurements of ν for U^{235} and Pu^{239} as functions of neutron energy, determined by measurements in a liquid scintillator tank. He reported the variation of ν to be approximately linear with energy in the Mev region, with a slope of about 1 neutron per 7 Mev for U^{235} and 1 neutron per 8 or 9 Mev for

Pu^{239}; these slopes appear to be relatively constant up to 14 Mev. He also reported that recent measurements by Hansen and Keepin gave ν values in the high-energy ranges approximately 3 per cent less than the currently accepted values. The values for U^{235}, if extrapolated to thermal energy, would give $\nu = 2.42 \pm 0.02$; the corresponding thermal value for Pu^{239} (normalized to 2.42 for U^{235}) would be 2.84 to 2.94.

Cranberg[13] discussed measurements of the inelastic scattering cross section of U^{238} by the time-of-flight method. The results at an energy of about 1 Mev are considerably lower than previous measurements.

The estimation of average fission-product cross sections by theoretical principles was discussed by Moldauer.[14] The analysis indicates about 10 barns per fission at 0.1 kev, 1.5 barns at 1 kev, 0.3 barn at 10 kev, and 0.1 barn at 100 kev.

Okrent[15] discussed (1) the range of breeding ratios that might be expected for fast breeders of various compositions and (2) the effects of uncertainties in the basic experimental data on these calculated breeding ratios. For a large reactor (1500-liter core) containing 20 per cent steel, 20 per cent sodium by volume, with the remainder fuel, he estimated the breeding ratio ranges: 1.6 to 1.7 for metal fuel, 1.3 to 1.45 for oxide fuel, approximately 1.35 for carbide fuel, and approximately 0.93 for iron diluent. The uncertainties in breeding ratio due to uncertainties in α, scattering and absorption cross sections, and fission-product cross sections are not so large as to be important in the feasibility of breeding. For example, if the maximum reasonable spread is assumed for α (the capture-to-fission ratio) the results are:

Fuel	Breeding ratio with		
	Probable α	High α	Low α
Pu (metal)	1.62	1.52	1.78
Pu (oxide)	1.32	1.22	1.50

Changes in composition could cause larger effects on breeding ratio. The use of a high percentage of molybdenum in the composition would, for example, produce a considerable reduction in the breeding ratio, and the use of tantalum rather than steel was stated to cause a very large decrease in breeding ratio in typical large solid-fuel breeders, which are characterized by relatively high dilution of the fissionable isotope.

In view of the difficulties that have been encountered in developing a metallic fuel for the fast breeder, studies of the performance of breeders using nonmetallic fuel are of particular interest. Greebler[16] discussed an oxide-fuel reactor which was fed with a mixture of U^{238} and plutonium of isotopic content typical of that produced by long irradiation in a thermal reactor. The composition of the reactor was, by volume, 50 per cent fuel oxides, 20 per cent steel, and 30 per cent sodium. In a 300-Mw(e) reactor using oxides of 90 per cent theoretical density, the calculated breeding ratio was 1.25, of which 0.7 was internal. The minimum doubling time was estimated as approximately 17 years. A fuel life of 100,000 Mwd/ton was assumed, and it was stated that tests had been run to approximately one-half this value. It was also stated that, if a large fraction of the fission products were lost by the high-density fuel, it might be necessary to go to oxide of lower density (e.g., 65 per cent of theoretical) in order to provide volume for the fission-product gases. In this case the breeding ratio was reduced, and the doubling time increased to about 22 years.

Another fuel variation was discussed by Nims,[17] a dispersion of plutonium carbide in uranium–15 wt.% molybdenum. It was stated as a more or less general principle that the most economic reactor design is one which does not strive for extremely high power density; the reactor considered produced 800 Mw(t) in a 1750-liter core, giving a specific power of 0.83 Mw(t) per kilogram of Pu^{239}. The calculated breeding ratio was 1.58. If the reactor were operated as a U^{233} breeder, it would use fuel of U^{233}-carbon in thorium. In this case, the specific power was a bit lower, 0.80 Mw(t) per kilogram of U^{233}, and the calculated breeding ratio was 1.26.

Thermal Breeders

Breeding in the thermal or near-thermal neutron-energy range appears to depend on the use of U^{233} as fuel. Physicswise it is much easier in fluid-fuel reactors than in solid-fuel reactors, if indeed it is feasible at all in the latter type. The main advantages of the fluid-fuel types in this respect are the elimination of neutron losses to fission products, the reduc-

tion of parasitic absorption by reactor structure, and the minimization of losses by neutron absorption in protactinium.

Following the 1958 Geneva Conference, there was considerable doubt as to the value of η (neutrons produced per neutron absorbed) of U^{233}; certain British measurements had indicated that it might be considerably lower than the value 2.28 which had previously been accepted [reviewed in *Power Reactor Technology*, 2(2): 16-17 (March 1959)]. At the Argonne conference, recent U. S. data were presented[18] which support a value at least as high as 2.28, and possibly slightly higher. More recent British data were also presented by Sanders.[19] Errors have been found in the original sample analyses which were used in the low η measurements. The current British program is closely coordinated with that in the United States, and samples of U^{233} are exchanged for measurement. Although this program is not yet complete, the British results up to now cannot be said to confirm those in the United States, nor can they presently be said to contradict them. Chernick[20] summarized the recent U. S. data and interpreted them in terms of reactor applications.

In summarizing the thermal-neutron measurements, which were made both by means of critical experiments, using U^{233} as the fuel, and by means of activation of a manganese bath with neutron beams, Chernick concluded that the thermal value is at least as high as 2.28 and that there is little evidence of important variation over the thermal part of the neutron spectrum. However, at energies above 1 ev, there is considerable fluctuation, with rather severe dips in η in the low-lying resonances (e.g., those at 2.3 and 6.8 ev). These effects will decrease the average value of η in most reactors employing practical moderator/fuel ratios. Chernick has derived average values of η as a function of the moderating ratio $\xi\Sigma_s/N_{23}$. As this ratio is increased from a value of 100, $\overline{\eta}$ decreases from 2.20 to a minimum of 2.16 at $\xi\Sigma_s/N_{23}$ of approximately 300. With further increases in moderation, $\overline{\eta}$ increases to substantially 2.28 at $\xi\Sigma_s/N_{23} = 10{,}000$.

Obviously, since the loss of neutrons by absorption in the moderator increases with an increase in the ratio $\xi\Sigma_s/N_{23}$, there will be an optimum ratio which gives the highest potential breeding ratio for any given moderator, and this ratio will be higher, the higher is the moderating ratio $\xi\sigma_s/\sigma_a$ for the moderator. Chernick finds this optimum ratio to be near 10,000 for D_2O and near 2000 for graphite, beryllium, and H_2O. The corresponding values of breeding ratio, if there were no neutron losses other than those to the moderator, would be 1.26, 1.23, 1.22, and 1.20. From these rather idealized considerations, it would appear that breeding is potentially possible with any of these moderators.

When beryllium is used as the moderator, there may be a significant further contribution of excess neutrons by the $(n,2n)$ reaction in beryllium. This is partly counteracted by the (n,α) reaction which absorbs neutrons parasitically and produces Li^6; Li^6, itself a strong neutron absorber, upon absorbing a neutron forms tritium, which in turn decays (12.5-year half life) to He^3, also a strong absorber. Chernick, from calculations based on the few $(n,2n)$ cross-section measurements in existence, concludes that the net direct effect of the $(n,2n)$ and (n,α) reactions is to give an apparent fast effect, ϵ, between 1.06 and 1.075. The Li^6 produced, he stated, has the effect of an additional stable fission product having a yield of up to 4 per cent and a cross section of 1000 barns. Häfele,[21,22] at the same conference, reported estimates of the net ϵ due to the $(n,2n)$ and (n,α) reactions of 1.051 to 1.076 for beryllium metal and 1.027 to 1.046 for BeO. If Li^6 is left in the beryllium until it approaches the equilibrium concentration, a poisoning, $p\alpha$ (neutrons absorbed in Li^6 per fission neutron produced) results, which, Haefele estimated, reduces the net "fast effect" to $\epsilon - p\alpha = 1.009$ to 1.034 for beryllium metal and $\epsilon - p\alpha = 0.998$ to 1.017 for BeO. Chernick suggested that beryllium be reprocessed periodically to remove Li^6.

At the Argonne conference, representative designs of the three major types of fluid-fuel thermal breeders were discussed: the aqueous homogeneous reactor,[23] the molten-salt reactor,[24] and the liquid-metal-fuel reactor.[25] Consistently with the purpose of the conference and with the recent reorientation of the fluid-fuel reactor program toward the long-range development of breeders, these designs emphasized the breeding potential. The aqueous homogeneous reactor discussed was a two-region reactor using a solution of UO_2SO_4 in the core and a ThO_2-D_2O slurry as the blanket. A breeding ratio of about 1.09 was estimated. Attention was called to the effect of corrosion products on breeding ratio and to the necessity for mini-

mizing this effect. The molten-salt reactor and the liquid-metal-fuel reactor designs described made considerably larger departures from past concepts in order to improve their breeding characteristics. The important new feature of the molten-salt design was the use of a reentrant graphite fuel tube for the fluoride fuel solution. The new features of the liquid-metal-fuel concept were a close-coupled design, in which the tantalum primary heat exchanger was located immediately adjacent to the upper blanket in order to minimize fuel inventory, and a new design of the all-graphite core structure to minimize sealing problems. The liquid-metal-fuel design was a two-region reactor with a solution of uranium in bismuth in the core and a ThO_2 suspension in bismuth for the blanket. Doubling times of 10 to 18 years were calculated for the molten-salt reactor and 6.8 to 10.6 years for the liquid-metal-fuel reactor.

In considering the breeding potentialities of thermal solid-fuel reactors, the loss of neutrons to fission products is highly important. Analyses of the existing experimental cross-section data were reported by Pattenden.[26,27] The fission products can be divided into two groups: the high-cross-section products having cross sections greater than 10,000 barns (Xe^{135}, Sm^{149}, Sm^{151}, Eu^{155}, Cd^{113}, and Gd^{157}) and the low-cross-section products having cross sections less than about 500 barns. The high-cross-section fission products reach a saturation concentration after an irradiation of the fuel to about 10^{20} neutrons/cm^2. In a high-flux reactor their saturation poisoning (the poisoning is defined as the number of neutrons absorbed per fission) is just equal to their yield; the combined effect is about 0.078. The low-cross-section poisons correspond roughly to an absorption cross section of 46 to 68 barns per fission, but the neutron spectrum must be taken into account in arriving at an effective thermal cross section. Pattenden discussed spectral effects and collected the existing data on resonance integrals of the fission products.

More recently, reactivity measurements on Shippingport type blanket rods irradiated in the MTR to exposures of 6000 to 13,000 Mwd/ton have been reported.[28] The variation in reactivity with exposure is stated to be consistent with a value of about 50 barns per fission for the poisoning effect of the long-lived fission products.

The solid-fuel systems discussed at the Argonne conference were of three types: a graphite-moderated helium-cooled reactor,[29] a "moderator-controlled" pressurized-water reactor,[30] and a "mixed-spectrum" sodium-cooled D_2O-moderated reactor.[31] The gas-cooled reactor is of the two-region type, with a curtain of moderator between the core and the blanket for reactivity control. A high specific power is used in the core to minimize the doubling time, and the two-region arrangement is used to reduce the neutron losses to protactinium. The moderator-controlled pressurized-water reactor is of the type which has been referred to elsewhere as a reactor controlled by spectral shift.[32–34] The moderator coolant is a mixture of D_2O and H_2O, and the ratio of the two materials is changed to vary the resonance absorption and thus compensate for reactivity changes. This arrangement improves the neutron economy by avoiding neutron losses to parasitic control rods and by reducing the absorption losses in H_2O. The mixed-spectrum reactor* employs very large sodium-cooled fuel clusters in the D_2O moderator. A large fast effect is attained by using plutonium-U^{238} as fuel, and losses of thermal neutrons are held low by using the D_2O moderator. This reactor might be considered a special type of coupled fast-thermal system, which was discussed in its more usual form at the conference by Avery.[36]

Although the solid-fuel reactors described may have the potentiality of breeding gains greater than zero, it appears that the designer may find the demands of low neutron loss so difficult to meet that the reactors will be penalized economically. It seems probable that if thermal solid-fuel breeders are eventually developed, they will evolve gradually from nonbreeders of high conversion ratio as the costs of natural fissionable isotope become high enough to justify the greater difficulty of the breeder design.

Other Information

Other papers presented at the Conference on the Physics of Breeding are listed as refer-

*This reactor was also discussed by Wigner in the first Annual Distinguished Lecturer's Address at the June 15, 1959, meeting of the American Nuclear Society.[35] See also the discussion of Sodium Deuterium Reactors elsewhere in this issue.

ences 37 to 45. The proceedings of the conference are being prepared for publication by Argonne National Laboratory. A recent bibliography on breeding is cited as reference 46. The papers presented at the thorium-U^{233} symposium[47] in January 1958 are pertinent to considerations of thermal and near-thermal breeders. Recent work on neutron diffusion and slowing down in beryllium is reported in references 48 and 49.

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Heavy-water Lattices

Buckling measurements have been performed on a series of approximately 250 natural uranium–D_2O test lattices, which were studied in the central test region of the Process Development Pile (PDP) at the Savannah River Laboratory (SRL). The results have been published in various SRL reports during the course of the work. A review of the experimental data was given in a paper[1] presented at the Second United Nations International Conference on the Peaceful Uses of Atomic Energy. A recently published supplement[2] to this review paper gives further information on the measurements, the method of data interpretation by which the SRL recipe for calculation of lattice parameters is evolved, and the possible sources of errors in the measurements.

A discussion[2] of the SRL experience compares the critical and subcritical buckling measurements on identical lattices. Some observers share the opinion that there is a single systematic difference between buckling measurements made in critical and in subcritical experiments. The experience of the SRL investigators does not support this opinion. Their study indicates that the spread in values reported for similar lattices by the various critical and exponential facilities shows far larger variation between similar facilities than any average difference between the two types.

The SRL recipe was used to compare the PDP measurements with similar buckling measurements on clustered rods and plates made at Chalk River and Saclay. The agreement is extremely good for clusters of UO_2 rods. The agreement in the region of maximum buckling is also satisfactory for clusters of metal rods and plates; however, the PDP measurements are consistently higher than those of Saclay at high moderator to fuel volume ratios.

It was found that the SRL recipe predicts consistently higher bucklings than those measured at Chalk River and Saclay for assemblies consisting of single rods. No PDP measurements on uniformly spaced rods were included in the formulation of the SRL recipe. To track down this discrepancy, measurements have recently been made on lattices of uniformly spaced small-diameter rods (0.50 and 0.69 in. in diameter); the measurements displayed a similar disagreement with the recipe. This suggests a partial breakdown of the recipe, which was derived for clustered assemblies, in the extension to the case of single-rod assemblies. The evidence indicates a systematic error in the PDP measurements that becomes appreciable for lattices with high values of thermal diffusion area. The effect arises because of the use of separated fuel and poison rods in the lattice of the outer region (i.e., the region around the central test region in the two-region assembly used for the measurements).

In support of the plutonium recycle program, the multiplying properties of several D_2O-moderated lattices were investigated in the Physical Constants Testing Reactor (PCTR), and a correlation of the data obtained is presented in reference 3. Four fuel clusters were investigated, each at three lattice spacings. Two of the clusters contained 19 natural UO_2 rods, one with air and the second with D_2O "coolant." The other two clusters were formed by replacement of three of the UO_2 rods with plutonium-aluminum alloy rods.

The measurements in the PCTR give k_∞ and f, the thermal utilization. From the values of k_∞ and f, as measured at the three lattice spacings, values of $\eta\epsilon$ and effective resonance integral were deduced in reference 3. The assumption made is that η is independent of changes in neutron spectrum over the range of fuel spacings and that ϵ is a function only of the fuel-element geometry. The parameters deduced in reference 3 are given with similar results from measurements at other laboratories.

Activation techniques[4] in the PCTR give $p = 0.868 \pm 0.0087$ and $\epsilon = 1.0178 \pm 0.0017$ for the D_2O-cooled cluster of 19 UO_2 rods in the 8-in. lattice. The correlation of reference 3 gives $p = 0.873 \pm 0.014$ for this case and, if the activa-

tion measurement of fast effect is used, results in $\eta = 1.321 \pm 0.024$. A "best" value of η for natural UO_2 is 1.322 ± 0.021 on the basis of the air-cooled and D_2O-cooled cluster data.

The methods of Carlvik and Pershagen[5] were used to calculate effective surfaces for the clusters; Hellstrand's[6] data were then used for calculation of effective resonance integrals. The calculated resonance integrals are compared with those inferred from the experiments in reference 3. The comparison is given as follows:

Coolant	Fitted RI	Hellstrand RI
D_2O	14.5	13.0
Air	12.4	12.1

Satisfactory agreement is indicated between the Chalk River critical measurements on D_2O-moderated clusters of 19 UO_2 rods and the k_∞ measurements in the PCTR.

Some analysis is made in reference 3 of the data obtained for clusters of 16 UO_2 and 3 plutonium-aluminum rods. To account for the epithermal absorptions (and fissions) in the plutonium resonances, the usual 4-factor formula is replaced by

$$k_\infty = \frac{\eta \epsilon f p^{28} p^{49}}{1 - \epsilon \eta_r^{49} p^{28} (1 - p^{49})}$$

where 28 refers to U^{238}, 49 refers to Pu^{239}, η_r is in the resonance region, and η is the average number of fission neutrons per thermal absorption in all the fuel.

From a correlation of the same type as that used for the 19 UO_2-rod clusters, a value of $\eta = 1.505 \pm 0.010$ is deduced for the particular UO_2 and plutonium-aluminum combination used in the experimental cluster. Considering the inaccuracies involved, no attempt was made to derive values of η or η_r for plutonium from the experiments.

Measurement of Doppler Coefficient

The techniques developed for measurement of k_∞ of a heterogenous reactor lattice with the PCTR can be extended[7] to measurement of the temperature coefficient of k_∞ with a change in temperature of the fuel. Measurements of the fuel temperature coefficient are reported[8] for the 19-rod cluster of natural UO_2 with air coolant within the fuel assembly using a D_2O-moderated 8-in. triangular lattice.

The fuel rods in the central test lattice were heated with a 10-mil-diameter Nichrome heating element that was covered with alumina and woven in the interelement space between the middle and outer rings of fuel rods. The fuel-rod cluster was contained within a $\frac{1}{8}$-in.-thick aluminum process tube. In turn, the process tube was centered within a $\frac{1}{8}$-in.-thick aluminum shroud tube, and there was a $\frac{1}{8}$-in. annular gas gap between the process tube and the shroud tube. This annular gap was filled with lampblack to reduce heat leakage to the moderator. Iron-constantan thermocouples were used in the test section to determine the temperature variation in the fuel cluster and an average fuel temperature. The temperature of the moderator associated with the central test lattice cell was also recorded at all fuel temperature readings.

Heating of the test section resulted in a decrease in reactivity of the PCTR, as determined by gradual withdrawal of a calibrated control rod to maintain criticality. After appropriate corrections the change in PCTR reactivity was plotted against change in fuel temperature; the fuel temperature range covered in the experiments is 30 to 365°C. Two sets of data are given on a graph in reference 8, one with the lampblack insulation present and the other with the insulation removed. By removal of the lampblack insulation, the heat leakage to the moderator was increased by a factor of 3. However, the two sets of data display essentially the same negative slope of PCTR reactivity versus fuel temperature. This is taken as proof that, in the experiments, only the resonance escape probability p of the lattice is changing with fuel temperature.

The experimental results give

$$\frac{1}{p}\frac{\Delta p}{\Delta T} = \frac{1}{k_\infty}\frac{\Delta k_\infty}{\Delta T} = -(2.3 \pm 0.2) \times 10^{-5}/°C$$

The quoted error is considered to be a maximum value.

The change in effective resonance integral of the UO_2 with temperature may be inferred from the above results by using the k_∞ and f measurements previously made at several lattice spacings. This approach yields

$$\frac{1}{\Sigma_0}\frac{\Delta\Sigma}{\Delta T} = (2.0 \pm 0.2) \times 10^{-4}/^{\circ}\mathrm{C}$$

where Σ_0 is the effective resonance integral at 20°C. This coefficient includes a small effect due to the thermal expansion of the aluminum spiders holding the rod cluster, which increases the effective surface to mass ratio of the fuel. This effect has been calculated to reduce the above value by 5 per cent so that the true measured Doppler coefficient is $(1.9 \pm 0.2) \times 10^{-4}/^{\circ}\mathrm{C}$ for a cluster of fixed separation.

It is of interest to compare the value of the Doppler coefficient obtained from the PCTR experiments with that indicated by the fairly recent measurements of reference 9 which utilized the activation technique. The work of reference 9, which was reviewed previously,[10] covers a range of surface to mass ratios and a range of temperatures up to 1000°C for UO_2. For the surface to mass ratio of the 19 UO_2-rod cluster investigated in the PCTR, the experimental results of reference 9 indicate a Doppler coefficient of $0.7 \times 10^{-4}/^{\circ}\mathrm{C}$. The discrepancy with the PCTR measurement is large and represents an important uncertainty in design data, particularly for gas-cooled reactors in which the Doppler coefficient is relied on to contribute the important negative power coefficient of reactivity. Further investigation of the possible errors and extraneous effects introduced in measurement of the Doppler coefficient appears necessary before there is confidence in the precision of measured values.

Thermal-neutron Flux Distributions in Fuel Specimens

Results of a systematic experimental investigation have been reported[11] for the radial distribution of thermal neutrons in cylindrical fuel specimens during irradiation. The investigation was initiated to promote more accurate definition of the conditions under which test specimens are irradiated, e.g., the distribution of fission events and fission damage and the calculation of temperature distribution within a cylindrical sample. Bars of natural uranium and of magnesium-cadmium alloys were used. The latter were employed to simulate the absorption properties of fuel materials of various enrichments. The specimens were activated in the almost pure thermal flux of the reflector of the CP-5 reactor.

Effects of the variable cylinder radius, a, and macroscopic absorption cross section, Σ_{abs}, in the range $0 < a\Sigma_{abs} < 2.2$ were studied. Within this range, the experimentally determined radial distribution of the thermal-neutron flux could be quite accurately fitted by an empirical formula involving empirically determined constants and the variable $a\Sigma_{abs}$.

The experiments were performed with great care, and a detailed tabulation is given of the experimental data, together with quite precise specifications of the specimen composition. The data should be of interest in checking out formulations for rod disadvantage-factor calculations.

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ETR Hot-channel Hot-spot Factors

The thermal design of power reactors is influenced to a large extent by the concept of the hot-channel factor. References 1 to 3, although they apply to a test reactor, illustrate the detail that is necessary in evaluating individual hot-channel factors, particularly in the area of the effect of flow distribution, when fuel-element wall temperature is a limiting factor on power rating. Reference 1 presents hot-channel hot-spot factors for Engineering Test Reactor (ETR) fuel elements operated at 175 Mw(t); these fuel elements are of the pinned type with equal spacing between fuel plates and solid side plates. Reference 3 presents more up-to-date information on ETR hot-channel factors and sample calculations on how the hot-channel factors can be utilized. The revised hot-channel hot-spot factors are presented in Table VI-1. The defining equation for use of the hot-channel hot-spot factors is as follows:

Maximum wall temperature = t_{in} +

$$F_T + F_{\Delta T}\Delta t_b + \frac{F_{q/A}\,(q/A)}{[(VF_v)^{0.8}/d_e^{0.2}]} \times b'$$

where t_{in} = inlet water temperature, °F
Δt_b = nominal bulk-water-temperature rise, °F
q/A = heat flux, Btu/(hr)(sq ft)
V = coolant velocity, ft/sec
d_e = minimum hydraulic diameter, ft
b' = 0.8 × b in the equation $h = bV^{0.8}/d_e^{0.2}$ (the value 0.8 is the uncertainty associated with the correlation equation)
b = parameter (temperature dependent) in correlation equation

It is evident from Table VI-1 that the largest hot-channel factor has to do with the channel-to-channel variation in coolant velocity. The heat-flux horizontal distribution factor is determined mainly by flux peaking due to an adjacent shim-rod guide tube. The variation in coolant velocities in various ETR experimental fuel elements has been reported in reference 2, and Table VI-2 illustrates the advantages accruing from the application of detailed analytical and experimental investigations to the fuel-element design.

The fuel element with staggered fuel plates is an element wherein the coolant-channel thickness is not uniform; the thickness is about 0.105 in. at the center of the element, increasing to about 0.120 in. for the outermost channels. It is planned to operate the ETR in the future with fuel elements similar to this test element.[2] Although cost is not the primary consideration in the design of a test reactor, the resulting 12 per cent reduction in total reactor flow and 17 per cent reduction in core pressure drop do have

Table VI-1 ETR HOT-CHANNEL HOT-SPOT FACTORS[3]

Hot-channel Factors	
Fuel content per plate	1.02
Heat-flux horizontal distribution	1.15
Power measurement error	1.05
Channel-to-channel variation in velocity	1.18
Variation of hydraulic diameter effect upon velocity	1.05
Variation of channel cross-sectional area	1.10
Total $F_{\Delta t}$	1.68
Additive inlet-water-temperature measurement, F_T	3°F
Heat-flux Factors	
Variation of fuel-core-alloy area	1.08
Heat-flux horizontal distribution	1.15
Fuel content per plate	1.02
Fuel-core-alloy-thickness	1.10
Net heat generated in meat and cladding	0.90
Power measurement error	1.05
Total $F_{q/A}$	1.32
Heat-transfer Coefficient Factors	
Variation of velocity, F_v	0.71
Correlation equation	0.80
Additive pressure measurement error	6°F

Table VI-2 COOLANT-FLOW REQUIREMENTS FOR ETR WITH DIFFERENT TYPES OF FUEL ELEMENTS[2]

(Average Coolant Temperature = 120°F; 49 Fuel Elements in Core)

Description of fuel element	Average coolant velocity required, ft/sec	Core pressure drop required, psi	Total reactor flow, gal/min	Maximum lateral pressure across fuel plates, psi
Original design of "pinned" fuel plates	35	53	56,000	3.4
Slotted-side-plate fuel elements with $3/16$- by 2-in. slots	35	56	57,000	<1.0
Fuel element with modified end box	33	44	51,500	2.3
Fuel element with staggered fuel plates	30	44	49,000	1.3

significance relative to construction and operating costs as well as to reactor performance.

Between references 1 and 3 there was a change in the design principle used in applying the hot-channel hot-spot factors. Quoting reference 1:

> The basic criterion established was that the calculated wall temperature at the hot spot be equal to or less than the saturation temperature at the hot-spot pressure.

Quoting reference 3:

> The method used to calculate the maximum wall temperature in the hot channel consists of applying a number of factors to the nominal conditions and accumulating these factors. Experimental measurements, fuel-element specifications, and some engineering judgment are used in determining the magnitude of each of the factors. It appears unlikely that all of the factors enumerated will occur in one fuel element in the hot position in the reactor, but the probability of such an occurrence is unknown and the hot-spot hot-channel analysis results in some conservatism. For this reason, a given situation is considered tenable even though the saturation temperature at the hot spot in the channel may be exceeded.

The maximum wall temperature at the hot spot is calculated to be about 30°F above the saturation temperature.[3]

Additional references on the use of hot-channel factors are found at the end of this section.[4–6]

Boiling-water Heat Transfer

The September 1959 issue of this journal contained a discussion of critical heat flux and burn-out in water at high pressure. Reference 7 presents experimental information on the effect of surface roughening on burn-out heat flux for water at relatively low pressures.

Factors other than burn-out often limit the thermal performance of the power reactors designed; however, these limitations are subject to improvement through developments in fuel-element materials, fuel-element design, and greater subdivision of the fuel. Accordingly, the interest in ascertaining the values of critical heat flux, as a function of the many system variables, remains high.

The basic piece of equipment used in the experiments described in the reference was a heat-transfer loop. This contained an electrically heated test specimen: a $\frac{1}{2}$-in., type 304 stainless-steel tube. The test specimen was centered in a precision-bore tube, and distilled water was pumped downward through the annulus, and through a heat exchanger, in a closed loop. While a constant power input was maintained to the tube, the coolant flow was reduced until burn-out occurred.

Three types of surface roughness on the test specimen were evaluated in the test loop: longitudinal grooves, diamond knurls, and circumferential threads. The system pressure was about 50 psia, the coolant velocity was from 10 to about 30 ft/sec, and the degree of subcooling was from 18 to about 80°C. Pressure-drop measurements were also taken.

The results of the experiments are summarized in Table VI-3. The last column in the table illustrates that rough surfaces permit substantial increases in the burn-out heat flux without increasing the pumping power, at least within the ranges of variables studied. Difficulty with the longitudinal-grooved tube was experienced during the test in the form of severe vibration of the tube.

The method of roughening the tubes does not appear to be very difficult, and it could perhaps

Table VI-3 PERFORMANCE OF ROUGH SURFACES RELATIVE TO SMOOTH SURFACES[7]

Tube	Surface treatment	Ratio of burn-out heat flux of a rough tube to that of a smooth tube*†	Ratio of friction factor of a rough tube to that of a smooth tube†‡	Ratio of the burn-out heat flux of a rough tube to that of a smooth tube§
	Smooth	1	1	1
2	Coarse diamond knurl, 0.008 in. deep	1.52	1.56	1.42
6	Fine diamond knurl, 0.005 in. deep	1.42	1.63	1.32
7	Fine diamond knurl, 0.008 in. deep	1.58	2.03	1.42
1	Fine diamond knurl, 0.013 in. deep	1.80–2.10	2.78	1.55–1.80
5	Longitudinal knurl, 0.008 in. deep	1.30–1.40	1.58	1.21–1.30
4	30 threads per inch, 0.006 in. deep	1.64	1.56	1.53
3	64 threads per inch, 0.009 in. deep	1.77	1.66	1.64

*Compared at same velocity, pressure, and subcooling.

†Results do not account for any reduction in burn-out heat flux or friction factor caused by changes in equivalent diameter required to maintain the same pressure drop.

‡Compared at a Reynolds number of 62,000 for isothermal flow through annuli.

§Compared at same flow, head loss, subcooling, and pressure.

be applied, if warranted, to power-reactor fuel elements. Since power reactors usually operate at pressures considerably greater than 50 psia, data at higher pressures would be of interest. The reference is a first progress report, and this extension to higher pressures may be forthcoming. Kutateladze reports of increasing the critical heat flux some 30 per cent by roughening the surface of a graphite disk under test with a file.[8]

Reference 9 is an account of high-flux-boiling heat-transfer research done at the University of California Lawrence Radiation Laboratory. The apparatus consisted of an electron accelerator wherein the test specimen, a $\frac{1}{8}$-in.-thick by $6\frac{1}{4}$-in.-diameter copper plate, was bombarded by electrons on one side and contacted by cooling water on the other. A thermocouple was peened into the test plate to measure plate surface temperatures. Water velocity ranged from 2 to 41 ft/sec, water pressure ranged from 36 to 120 psia, and heat flux ranged to 15.2×10^6 Btu/(hr)(sq ft). Several different surfaces were employed for the tests: polished copper plate, as-rolled copper plate, sandblasted copper plate, solder-tinned copper plate, and gold- and nickel-plated copper plate.

The results of the experiments are quoted as follows:[9]

Burn-out flux is greatly affected by the surface condition of the plate. The highest burn-out fluxes achieved were with a copper plate buffed to a mirror finish. After operating for a long time and becoming oxidized, the burn-out flux of the polished plates became less and approached that of the "as-rolled" smooth copper plates. The lowest burn-out fluxes were obtained with an as-rolled copper plate with the surface "tinned" or coated with a very thin layer of ordinary "soft" solder (50-50 lead-tin); because the plate temperature at burn-out was well above the melting point of the coating, the tinned surface was liquid. This coating technique might be useful to isolate the influence of the surface finish from the other variables in boiling studies, by boiling from amorphous surfaces.

Other surface finishes used were: sandblasted, gold-plated, and nickel-plated. The sandblasted, gold-plated, and nickel-plated finishes gave only slightly higher burn-out fluxes than the tinned surface. The as-rolled plate burn-out fluxes were higher than those of the tinned plate but lower than those of the polished plates. All observed burn-out fluxes, at any given water velocity, fell between the polished surfaces on the high side and the tinned surface on the low side; the approximate range is from 8.3 to 25 kw/sq in. at 6 ft/sec and from 18 to 30 kw/sq in. at 35 ft/sec. Burn-out flux is proportional to v^m, where m is approximately equal to 0.12 for highly polished plates and 0.5 for the tinned plate.

The experimental data were also plotted using the Griffith correlation. The lowest values for burn-out heat flux (those for the tinned plate) were found to be from two to five times higher than the highest values predicted by Griffith's correlation. Although the authors do not attempt to explain this difference in detail, they do note that their $\frac{1}{8}$-in.-thick samples are considerably thicker than the heating surfaces reported in other high-flux nucleate boiling experiments and suggest that such a thick plate would have a much lower momentary temperature rise when a steam bubble passes over a point on the surface than would a thin plate.

The reference also contains a discussion of burn-out theory, which will not be summarized at this time. The paper concludes with a discussion of "boiling songs" observed during the tests:[9]

> During the experiment, audible, high-pitched sounds were produced in the apparatus. These sounds were not systematically investigated; however, they appear to be common to high-flux boiling and have been noticed in other laboratory apparatus such as accelerator targets and ion sources where high-flux forced-convection nucleate boiling occurs. The frequency of the sounds, which were relatively pure notes rather than noise, was measured for a few cases.
>
> A piezoelectric crystal, of the type used in earphones, was taped to the water-inlet hose just upstream from the high-voltage insulator and the signal was fed directly to an amplifier and oscilloscope. The frequency of the predominant note was photographed on the oscilloscope. The measured sound was transmitted through the apparatus and the water. Attenuation between the source and pickup, and the sound transmission properties of the apparatus, undoubtedly affected the character of the observed signal. Consequently only the predominant frequency, which was clearly displayed, was recorded. Frequency was higher for higher water velocities. The sound appeared at a heat flux just above the start of boiling. As the heat flux was increased, the sound amplitude also increased, but the frequency decreased. The background frequency, perhaps due to water cavitation, etc., from the apparatus, was about 2.5 kc/sec as measured with no heat flux. Frequency ranged from 20 kc, decreasing to about 4 kc as heat flux increased. In raising the heat flux to just below burn-out, the note fluctuated and disappeared at a plate temperature of approximately 420°F for 3.5 gal/min, 570°F for 9 gal/min, and 600°F for 18.6 gal/min, and a sporadic loud noise of predominantly random low frequencies (less than approximately 300 cycles/sec) appeared. At this point the visible character of the boiling was beginning to change with the formation of a "bubble" or steam film in the center of the $1\frac{1}{2}$-in.-diameter circle.
>
> No plausible theories have been offered in the literature as to the source of this sound. A previous investigation of such sounds in other experiments disclosed that no systematic study had been made, even to the extent of our rudimentary observations.

The source of the sounds was briefly discussed, but no clear evidence as to their origin was presented. Possibly useful information could be obtained by "listening" to water-cooled reactors.

Gaseous Coolants

Several reports have recently been issued, dealing with the transfer of heat to gases, wherein large film drops were utilized to study the influence of temperature-dependent properties on the heat-transfer coefficients. Reference 10 is an account of work done at Stanford University; after presenting analytical solutions for heat transfer to a fluid flowing in a circular tube, experimental data are presented and analyzed. The experimental data were taken with the following objectives:

1. To determine experimentally the effect of large temperature differences for a gas being heated at approximately constant heat rate per unit of tube length in laminar flow in a circular tube, and to determine if an entry-length solution of the Graetz type (based on constant properties) can be used, with suitable modifications if necessary, under large temperature difference conditions. This part of the investigation is concerned with local heat-transfer conductances measured along a tube.
2. To determine experimentally the effect of large temperature differences for a gas being cooled with a constant surface temperature in laminar flow in a circular tube. This part of the investigation is concerned with mean heat-transfer conductances with respect to tube length.
3. To determine experimentally the effect of large temperature differences for a gas being cooled with a constant surface temperature in turbulent flow in a circular tube. This part of the investigation is concerned with mean heat-transfer conductances with respect to tube length. Consideration of the data leads to the following conclusions: In the cases of laminar-

and turbulent-flow cooling, and laminar-flow heating "... the constant-property analytical solutions, or the small temperature difference experimental correlations, are believed to be of acceptable accuracy for engineering calculations if all fluid properties are evaluated at the local mixed-mean temperature where local Nusselt numbers are concerned, and the mean temperature with respect to length in the case of mean Nusselt numbers."[10] No experimental data were taken for the case of turbulent-flow heating, but the work of other experimenters was examined, and the following correlation was suggested: (Nusselt No., turbulent, heating) = (Nusselt No., turbulent, constant fluid properties) $(T_{\text{wall}}/T_{\text{mixed mean}})^{-0.5}$.

Reference 11 is a National Aeronautics and Space Administration (NASA) publication dealing with heat-transfer and friction coefficients for helium flowing in a tube under the following conditions:

Average surface temperature, 1457 to 4533°R
Average Reynolds number, 3230 to 60,000
Maximum heat flux, 345,000 Btu/(hr)(sq ft)
Maximum exit coolant speed, Mach 1.0
Pressure, near atmospheric

The electrically heated test section was fabricated of molybdenum and tungsten, with the molybdenum tube having a length to diameter ratio of 60, and the tungsten tube, 92. Since the tungsten test section was not instrumented with pressure taps, only the molybdenum tube was used to obtain pressure-drop data. The average* friction coefficients were calculated using two values of gas density, one at the average static pressure and temperature of the gas and the other at the average film temperature. The average* heat-transfer coefficients were calculated using the difference between the average tube-surface temperatures and average bulk-fluid temperatures. Local heat-transfer coefficients were also calculated using the conventional definition; bulk temperatures as functions of length were calculated by means of a heat balance around the heated lengths.

For a range of Reynolds numbers of from 10,000 to about 25,000, the data were found to agree with the following correlation to within 10 per cent:

$$(\text{Nu}_{\text{av}})_f = \frac{h_{\text{av}}\, d}{k_f} = 0.021\ \text{Re}_f^{0.8}\ \text{Pr}_f^{0.4}\left[1 + \left(\frac{l}{d}\right)^{-0.7}\right] \qquad (1)$$

All the physical properties and the density were evaluated at the film temperature. The symbols have their conventional definitions with l/d defined as the length to diameter ratio of the test section. Equation 1, with the properties evaluated at the surface temperature, was compared with the experimental data, and the authors conclude that "for the limited range of Reynolds number in this investigation, the average heat-transfer data seem to be correlated better by evaluating the physical properties and density at the surface temperature." Local heat-transfer coefficients were found to be in "fair" (±15 per cent) agreement with the following correlation:

$$\text{Nu}_f = \frac{hd}{k_f} = 0.021\ \text{Re}_f^{0.8}\ \text{Pr}_f^{0.4} \qquad (2)$$

When Eq. 2 was evaluated using properties taken at the surface temperature, the experimental values were about 20 per cent above the correlation. In general, the precision of the experimental points showed a spread of from 25 to about 40 per cent, depending on whether surface- or film-temperature properties were used.

Average friction coefficients for adiabatic runs were in good agreement with the Kármán-Nikuradse correlation

$$\frac{1}{\sqrt{8f/2}} = 2\ \log\ (\text{Re}\ \sqrt{8f/2}) - 0.8 \qquad (3)$$

Average friction coefficients for the runs in which heat was transferred were in "poor" agreement with Eq. 3 when the density was evaluated at the film temperature and in "fair" agreement when the density was evaluated at the gas temperature.

The symbols in the above equations have the following meanings:

Nu = Nusselt number
Re = Reynolds number
Pr = Prandtl number
f = friction factor
l/d = length to diameter ratio
Subscript f = properties evaluated at film temperature

*In this context the "average" value is the one based on consideration of the test section as a whole, rather than looking at one particular station or area of the test section.

Short Notes

Two recently published reports[12,13] should be useful in specialized areas of reactor design. Reference 12 is a collection of lecture notes on heat extraction from boiling-water power reactors; these notes were presented at the advanced summer institute at Kjellèr, Norway, in August 1959. The chapter headings are: Heat Transfer; Two-phase Pressure Drop; The Prediction of Steam Volume Fractions; Critical Heat Flux or Burn-out; Calculation Procedures for Boiling Systems; Thermal and Hydraulic Design Procedures for Boiling-water Reactors.

Reference 13 is an account of calculational techniques to aid in computing pressure drops and flow distribution within a pressurized-water reactor vessel. The main subjects covered are: Method and Equations for Pressure-drop Calculations; Method for Calculating Flow Distribution; Typical Examples of Flow-distribution Calculations; Formulas for Various Types of Pressure Drops; Application of Formulas to Reactor-vessel Pressure-drop Calculations.

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Prediction of the Neutron Distribution in the Reactor Shield

The design of a reactor shield depends to a large extent on the secondary gammas produced in the shield. In power-reactor shields the penetrating gammas are usually neutron-capture gammas born in the shield. The bulk of the shield heating is also neutron induced, especially beyond the thermal shield. For this reason the prediction of the neutron distribution in the shield is of utmost importance.

Simple diffusion or age-diffusion theory gives unsatisfactory results at large distances from the reactor core. The reason is that the theory neglects the long flights between collisions and the anisotropy of the scattered flux. In reference 1 is a discussion of several methods in various degrees of development, including their limitations. Of these methods, the S_n method seems to hold great promise. The most recent extension of the S_n method is presented in reference 2.

Another method for predicting the neutron-flux distribution combines neutron-removal theory with neutron diffusion. A two-group diffusion equation is developed in reference 1 for slab geometry. The fast-group source term is calculated using removal-cross-section methods which are also described in reference 1. Of interest is a comparison of the two-group method results with some ORNL Bulk Shielding Reactor (BSR) experimental data. The experimental configuration consisted of a single iron slab, approximately 6 in. thick, surrounded by water. In the vicinity of the iron slab, the two-group method results are in poor agreement, but the agreement becomes better as the distance from the iron slab is increased. Especially of interest is a three-group calculation (shown in the same figure) which shows better agreement.

The use of the combined removal and diffusion theory can describe the thermal-neutron distribution in a hydrogenous shield and, with proper selection of cross sections, holds promise in describing the thermal-neutron distribution in nonhydrogenous shields. A program recently coded on the IBM-704, called RE-34, seems to be ideal for use with the removal and source-diffusion theory.[3]

The shielding program is a one-dimensional, multigroup, multiregion code for slab, cylindrical, or spherical geometry. Supplied as input is the source distribution which is given as the highest energy group. Also available is a wide choice of boundary conditions. At the inner edge (the point of minimum r) the flux may be set equal to zero, the flux may be specified, or, if $r \neq 0$, the "black-boundary" condition,

$$-\frac{1}{\phi}\frac{d\phi}{dr} = \frac{1}{d}$$

may be imposed. At each interface the standard boundary condition of continuity of flux and current or the black boundary may be chosen. At the outer boundary the flux may be set equal to zero, the derivative of the flux may be set equal to zero, or the black-boundary condition may be applied. Reference 3 gives the description of the numerical method, the input format, the output format, and a print-out of the program.

Interaction of Fast Neutrons in Iron, Lead, Oxygen, and Nitrogen

Reference 4 is a report on the interaction of fast neutrons in nitrogen, oxygen, iron, and lead for various energies. The most important information to the shield designer is the inelastic neutron scatter gamma-ray production cross section as a function of the neutron-source energy. The angular distribution of the neutron inelastic scattered gammas was also investigated. The limited data for iron and lead show that the gamma distribution is anisotropic with the minimum yield at the laboratory angle of 90 deg. The angular distribution is symmetric about 90 deg. In both lead and iron the angular distribution of the gammas becomes less anisotropic with increasing neutron energy.

Gamma-ray Attenuation and Heating Codes

The calculation of the gamma-ray attenuation and heating in reactor shields by all computational techniques is, in general, very time consuming. For this reason codes for high-speed computers are of particular interest.

An IBM-704-709 program has been written by Atomics International for computation of the gamma-ray attenuation and heating.[5] It is a multigroup code for computing gamma-ray flux distributions in a multiregion semiinfinite or finite slab shield. As many as 30 regions and 20 energy groups may be included in a single calculation. In each region either a uniform or an exponential source distribution may be specified. Furthermore, by specifying a radius for a given source region, the program will assume a truncated source geometry for that region with the detector located at the apex of the cone. The build-up factor is represented by a double exponential expression of the form

$$B(\mu_r) = Ae^{-\alpha_1\mu_r} + (1 - A)\ e^{-\alpha_2\mu_r}$$

where μ_r is the penetration in mean free paths and A, α_1, and α_2 are energy-dependent material constants. One of the restrictions is that a single build-up factor is assigned for each region in which the flux distribution is desired. The build-up factor can be held constant or changed for each source region.

Included in the report are the basic equations, a print-out of the code, and the input and output formats.

Another multigroup and multiregion code for a slab source and shield has been programmed on the Ferranti Mercury Computer.[6] It is similar to the above code except that the source distribution is left as a variable and the source must be represented as an infinite plane.

Both programs are best suited for calculations on large reactors because of the slab-shield restriction. Also, no variation in the source intensity perpendicular to the source-detector center line is allowed. The choice of the build-up factor that will be representative for a multilayered shield will in some instances be difficult.

References

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7. New York Operations Office, June 12, 1959. (Unpublished.)

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Erratum

The reference cited in the shielding section of *Power Reactor Technology*, 3(1): 36 (December 1959) is not correct. The proper reference is listed as reference 7 at the end of this section.

Fabrication of UO_2 Fuel Elements

Uranium dioxide fuel has been rather generally adopted for civilian power reactors because it has the advantages of being relatively inert chemically and very resistant to radiation damage; its primary disadvantages are brittleness, susceptibility to thermal fracture, low thermal conductivity, relatively low density, and fission-gas release. Although the melting point of UO_2 is high, fuel elements of rather small diameter must be used if melting of the oxide at the center is to be avoided at reasonably high heat fluxes. This limitation on fuel-element size results in relatively high fabrication costs per unit mass of fuel because the number of manufacturing operations per unit mass of fuel is large, the problem of assembling the small elements into a complete reactor core is difficult, and the ratio of cladding material to fuel material is high.

Two methods are now under study for improving this situation: development of annular type fuel elements clad and cooled internally and externally; and modification of the fuel material to increase its conductivity. In addition, new fabricating techniques for producing UO_2 fuel elements may yield independent decreases in fuel costs.

Annular UO_2 Fuel Elements

Hanford[1] has reported on some of the preliminary irradiation tests performed on Zircaloy-clad UO_2 specimens, 1.44 in. OD, 0.395 in. ID, and 4.4 in. long. A disadvantage of the annular element was expected to be the expansion of the fuel material away from the unbonded cladding on the inner surface of the annulus and a consequent decrease in the cooling ability of that surface. However, examination of the specimens after very short irradiation showed that characteristic radial cracking of the fuel material occurred, and the gap which would have resulted from expansion of the continuous fuel was apparently eliminated by the reorientation of the radially cracked segments.

In selecting an oxide element for the Plutonium Recycle Test Reactor (PRTR), Hanford has developed a fuel assembly using coaxial annular fuel elements. The fuel assembly is 8 ft 4 in. long and consists of two coaxial Zircaloy-clad uranium oxide tubes and a central Zircaloy-clad rod. The outside diameter of the assembly is 3 in.; it is designed to fit into a $3\frac{1}{4}$-in.-ID Zircaloy vertical pressure tube.

In comparing the annular type element with a 19-rod bundle of $\frac{1}{2}$-in.-diameter rod type elements, Hanford[1] reported that the annular element contained 28 per cent more uranium per unit of length, had a higher fuel-to-cladding ratio, and had more uniform coolant channels. Physics data indicated that the annular element gave a higher reactivity, a smaller positive reactivity effect upon loss of coolant, a 2 per cent greater thermal utilization, and a 3 to 4 per cent higher conversion ratio.

Since the oxide fuel cylinders must still be rather short and must be manufactured to rather close tolerances to fit external and internal tubes which form the cladding, it is not expected that a large saving in fabrication cost would be obtained. In an integral 8-ft element of stacked UO_2 cylinders, ratcheting and thermal fracturing of the UO_2 may be problems. Hanford[1] has indicated that specimens 2 ft long are to be irradiated to check their performance.

Hot Isostatic Fabrication of UO_2 Fuel Elements

For the purpose of achieving fabrication-cost reductions, the AEC Division of Reactor Development has had an investigation made on various methods of fabricating UO_2 fuel elements, including the method of hot isostatic (H.I.) pressing. Sylvania-Corning Nuclear Corporation (Sylcor) conducted the investigation and has recently reported some of the data obtained.[2] The H.I. pressing method was applied to the fabrica-

tion of rod type elements clad with stainless steel.

Three fabrication techniques were studied, using sintered UO_2 pellets, green compacted pellets, and UO_2 powder.

Sintered UO_2 Pellet Elements. The UO_2 pellets used for this phase of the study met the following specifications: minimum density, 94 per cent of theoretical; diameter, 0.402 ± 0.002 in.; and height, 0.400 ± 0.015 in. Pellets having these specifications could be fabricated by cold compacting and sintering powders without grinding. These pellets were loaded into commercial stainless-steel tubing (0.010 to 0.015 in. thick) which would yield a minimum pellet clearance of 0.002 to 0.003 in. between the largest pellet and the smallest tube diameter. The H.I. pressing of the assembly produced some bonding of the pellet and cladding, and inspection of the assembly indicated that the interface between the cladding and the pellet was free of voids. Some cladding-surface defects were evident after H.I. pressing, including some pleating.

Green Compacted UO_2 Pellet Elements. The second phase investigated by Sylcor to obtain further saving in fabrication cost was the use of nonsintered and unground UO_2 pellets, H.I. pressed in commercial stainless-steel tubing.

In this phase, green UO_2 pellets of ceramic grade UO_2 were cold-pressed to 46 per cent of theoretical density, inserted in stainless-steel tubes, and H.I. pressed. The H.I. pressing densified the UO_2 to 91 per cent of theoretical density, but the large volumetric change caused pleating of the tube. To eliminate much of the pleating, Sylcor studied fabrication techniques for producing high-density green pellets. Results of this study were not satisfactory, and the H.I. pressing of elements using cold-pressed green compacts was discontinued.

Nonpelletized UO_2 Elements. Since the H.I. pressing of green UO_2 pellets proved to be unsatisfactory, Sylcor studied a third method of producing less costly fuel elements. This phase of their study consisted of tamping a blend of UO_2 powder directly into the stainless-steel cladding tubes at 40 tsi, in 3-g increments. After H.I. pressing, the specimen element had a UO_2 density of 95 to 98 per cent of the theoretical value, and an excellent mechanical bond between the UO_2 and the cladding was obtained.

It was concluded by Sylcor that the H.I. pressing of a tamped blend of UO_2 powder would have the following advantages over the present methods of making UO_2 pellet type elements:[2]

1. Handling of green compacts is eliminated.
2. Sintering of pellets is eliminated.
3. Grinding of pellets is eliminated.
4. Close-tolerance tubing is eliminated.
5. Inspection of pellets is eliminated.

An indication of the cost savings which may result is obtained by noting the costs, as reported by Sylcor, of pelletizing UO_2 ($6 to $8 per pound) and grinding UO_2 (about $1 per pound).

Of course, not all the above savings can be achieved since some cost must be attributed to the tamping and H.I. pressing process.

Although the H.I. pressing technique appears to show some promise of decreasing the UO_2 fuel-element costs, more effort appears to be required to achieve good tamping in long fuel elements and to maintain at the same time sufficient void in a nonactive part of the fuel-element assembly for fission-gas expansion. In addition, more data are required on the effect of H.I. pressing on the cladding material.

The development of new techniques for obtaining high tamping forces, combined with H.I. pressing of the assembled element, may lead to lower costs for rod type elements and appears to be particularly attractive for annular type fuel elements since the isostatic pressing should result in good mechanical bonding of both the internal and external cladding.

Irradiation of Cold Isostatic Pressed UO_2 Fuel Elements

Hanford[1] has prepared compartmented annular fuel elements by cold isostatic pressing of untamped UO_2 powder clad with aluminum–1 wt.% nickel alloy. In these fuel elements the aluminum cladding was allowed to collapse between the ribs forming the compartments of the annular extruded aluminum alloy tube. The loose UO_2 powder was compressed in the process to about 60 per cent of theoretical density. This type of fuel element, which contained 2.4 per cent enriched UO_2 and was 8 in. long by 1.4 in. OD, was irradiated in the MTR. The power generation of the element was calculated to be 70 watts per gram of UO_2, and the fuel temperature was estimated to be about 3815°F. Examination

after irradiation (no burn-up data available) showed no external dimensional changes; the UO_2 fuel had sintered and cracked, hollow cores developed in some of the compartments, and some shrinkage away from the compartment ribs occurred.

More irradiation data on elements prepared by cold and hot isostatic pressing will be needed to determine their performance relative to that of elements prepared by the conventional methods. The anticipated savings in fabrication costs, even if they amount to only a few per cent of the present total fabrication costs, are significant and should become more significant as the remainder of the total cost is reduced by other means.

Thermal Conductivity of UO_2 Fuel Elements

In UO_2 fuel elements of the present type, a significant fraction of the temperature difference between the cladding and the center of the fuel is thought to occur across the gap between the UO_2 body and the cladding. Consequently the thermal performance can probably be improved either by increasing the thermal conductivity of the UO_2 or by decreasing the temperature drop across the gap.

The latter improvement might be accomplished either by eliminating the gap between the UO_2 pellet and the cladding material or by using a bonding material of high thermal conductivity. The H.I. pressing of UO_2 fuel elements, discussed previously, eliminates the gap by essentially mechanically bonding the UO_2 to the cladding. Other approaches to elimination of the temperature drop across the gap are to use lead (Pb) as a bonding material, as was done in BORAX-IV, or by allowing the fuel cladding to collapse on the fuel under the reactor coolant pressure.

In-pile Measurements

Data presented by Westinghouse Atomic Power Division (WAPD)[3,4] on in-pile thermal-conductivity measurements of UO_2 with various diametral gas gaps indicated that the "effective" thermal conductivity of the UO_2 calculated from the measurements on the basis of a "smeared" model[5] (one in which the gap and the UO_2 are treated as though they were uniformly mixed) increased by a factor of 2 to 3 at low temperature when the diametral gas gap was reduced from 8 mils to zero. The low values of effective thermal conductivity were obtained on samples having 8-mil diametral gaps filled with xenon plus krypton. No data were presented to indicate whether cracking of the test specimen occurred which could materially change the effective thermal conductivity of the fuel element by changing the gap. A thermal conductivity of 2.58 to 3.72 Btu/(hr)(ft)(°F) was obtained for a UO_2 element having zero clearance between the UO_2 and the cladding.[4] This is consistent with the Hedge and Fieldhouse data for the thermal conductivity of UO_2 as shown in Fig. 1. Thermal-conductivity data from other sources are also shown in Fig. 1.

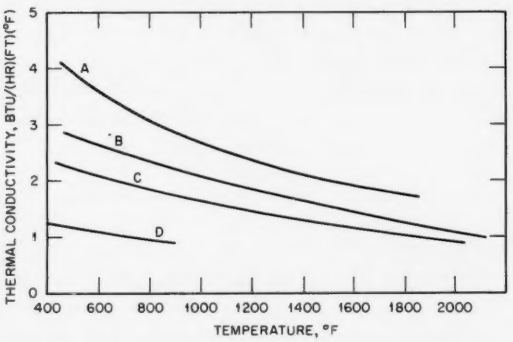

Figure 1 — Thermal conductivity of UO_2 as a function of temperature[5] (stoichiometric UO_2; k corrected to 95 per cent of theoretical density unless otherwise specified. *A*, Kingery data.[6] *B*, Hedge and Fieldhouse data.[7] *C*, Hedge and Fieldhouse data (75 per cent of theoretical density). *D*, Eichenberg data.[5]

The Kingery[6] and Hedge and Fieldhouse[7] data were obtained on out-of-pile specimens; the in-pile data were estimated by WAPD using a method described in reference 5.

Data are given in Fig. 2 for in-pile thermal conductivity versus center temperature as obtained by WAPD[5] on UO_2 pellets (~95 per cent of theoretical density) having a 0.3535-in. outside diameter and an initial 3.5-mil diametral gap between UO_2 pellets and the stainless-steel cladding. These data indicated that the effective thermal conductivity of UO_2, when the center temperature was between 750 and 1100°F, was 1.0 ± 0.2 Btu/(hr)(ft)(°F). The computation of the effective thermal conductivity was based on the assumption that the diametral helium-filled

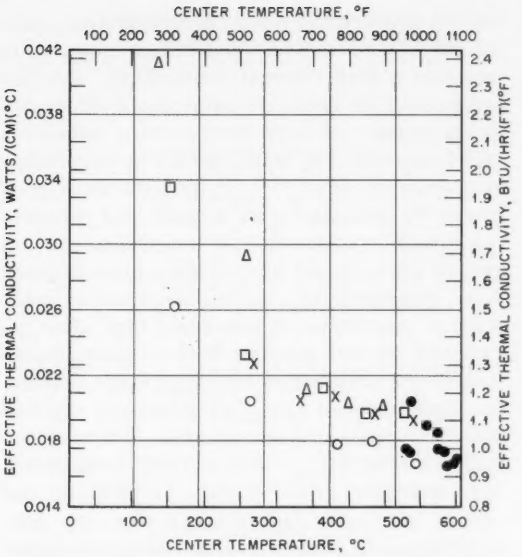

Figure 2—Effective thermal conductivity of 0.3535-in.-OD UO_2 pellets as a function of center temperature[5] (all values are calculated utilizing annulus Model 8 assuming 1 watt per gram of gamma heating in stainless-steel capsule). ○, Mar. 5, 1958, first start-up. □, Mar. 6, 1958, start-up. ×, Mar. 12, 1958, start-up. △, Mar. 18, 1958, start-up. ●, steady-state operation.

gap of 3.5 mils was maintained. It can be noted in Fig. 2 that the thermal-conductivity measurements made on the first reactor start-up were significantly lower than later measurements, especially at the lower center temperatures. WAPD[5] indicated that the large variation at the lower temperature may have been due to the measuring techniques, which are not very accurate at the lower heat-generation rates. However, since the model used to calculate the thermal conductivity assumed that the 3.5-mil diametral gas gap was maintained, part of the increase noted may also have been due to radial cracking of the fuel pellet followed by shifting of the fuel fragments to contact the cladding surface. Such radial cracking would have a relatively small effect on the thermal conductivity of the UO_2, but relocation of the fragments to close or reduce the diametral gas gap will cause a decrease of the temperature drop across the gap and thus allow the fuel to operate at a higher heat flux for a given central temperature. In a recent presentation on nuclear fuel elements,[8] O. J. C. Runnalls, head of the Fuel Development Branch, Atomic Energy of Canada, Ltd. (AECL), reported on tests made in the Hydraulic Rabbit[9] loop at Chalk River on 0.67-in.-OD oxide pellets operated at heat fluxes high enough to raise the center temperature to near the melting point. The tests indicated that a maximum pellet-surface temperature difference of ~212°F existed between fuel specimens having an initial diametral clearance of 0.017 in. and others having an initial diametral clearance of 0.005 in. It was further reported by Runnalls that J. A. L. Robertson of the AECL has suggested a more realistic model for calculation purposes. This model considers that the radially cracked segments of oxide shift radially outward to contact the cladding and that the interfacial temperature drop becomes mainly dependent on the interfacial pressure and on the properties of the contacting surfaces.

If further data substantiate this model, it would appear that larger clearances between fuel cladding and oxide pellets may be tolerated provided collapse of the cladding is not a problem.

Irradiation Effect

Other measurements, made by Ross of AECL,[10] of the thermal conductivity of UO_2 under different conditions were also presented by Runnalls.[8] Ross measured the thermal conductivity of stoichiometric UO_2 pellets at 140°F after irradiation to an integrated thermal-neutron flux of 2×10^{19} neutrons/cm^2. The maximum center temperature during irradiation did not exceed 932°F. A small (25 per cent) decrease in thermal conductivity was observed after an irradiation of 9×10^{17} neutrons/cm^2, and no further decrease was observed up to an irradiation of 2×10^{19} neutrons/cm^2. The absolute magnitude of the thermal conductivity as measured by Ross was 3.35 Btu/(hr)(°F)(ft) for the unirradiated specimen at 140°F and approximately 75 per cent of this value after an irradiation of 9×10^{17} neutrons/cm^2. These values are inconsistent with the data obtained in-pile by Eichenberg[5] but do appear to be consistent with the measurements of Hedge and Fieldhouse as shown in Fig. 1.

Effect of Oxygen to Uranium Ratio

Threefold lower values of thermal conductivity have been reported by Belle[11] and by Nichols[12] for uranium oxide pellets having an oxygen to uranium ratio 8 per cent above the stoichiometric UO_2. Measurements made by

Ross[10] on the variation of thermal conductivity of UO_2 at 140°F are shown in Fig. 3. These data appear to substantiate the reported observations of large decreases in thermal conductivity of UO_2 with increasing oxygen to uranium ratio. Some indication of the effect of

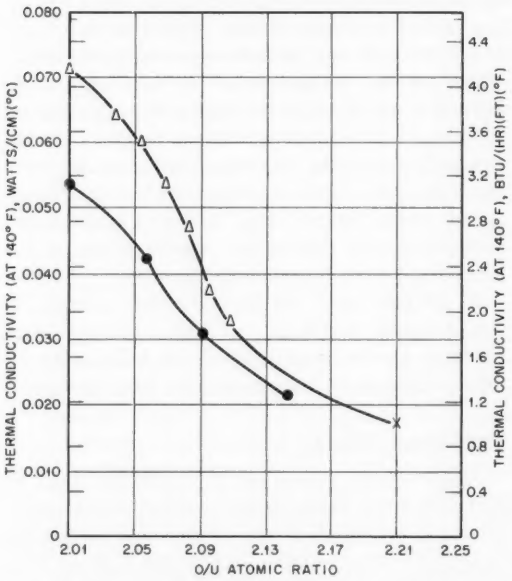

Figure 3—Thermal conductivity[8] of UO_2 as a function of oxygen to uranium ratio at 140°F. Δ, UO_2, initial density 10.3 g/cm^3, sintered in hydrogen at 1650°C, and oxidized at 900°C in 2 cm Hg of oxygen. ●, UO_2 + 0.1 per cent TiO_2, initial density 10.2 g/cm^3, sintered in hydrogen at 1650°C, and oxidized at 900°C in 2 cm Hg of oxygen. ×, UO_2, density 10.6 g/cm^3, sintered in steam at 1400°C, and cooled in steam.

higher oxygen to uranium ratio on the thermal conductivity of UO_2 at higher temperatures is evident from data reported recently by Sylcor.[13]

Two standard samples of $UO_{2.10}$ having a density of 10.99 g/cm^3 (98.5 per cent of theoretical density) and a test sample having the stoichiometric composition $UO_{2.00}$ and a density of 10.62 g/cm^3 (97 per cent of theoretical density) were compared. At 680°F the thermal conductivity of the $UO_{2.00}$ specimen was 25 per cent above that of the $UO_{2.10}$ specimen. Tests at a higher temperature, 790°F, showed a thermal conductivity of the $UO_{2.00}$ specimen to be 33 per cent greater than the $UO_{2.10}$ specimen. Although more data are needed to justify any assumption that the percentage difference in thermal conductivity of nonstoichiometric UO_2 increases with temperature, it appears to be quite evident that stoichiometric UO_2 should be used in fuel elements to maintain a high thermal conductivity. Further evidence of the decrease in thermal conductivity which occurs at high temperature with nonstoichiometric UO_2 was obtained by tests made by Robertson et al.[14,15] These data were presented by Runnalls[8] and indicate that nonstoichiometric uranium dioxide exhibits more grain growth and releases more fission-product gases than stoichiometric pellets irradiated under similar conditions. It was found that, after irradiation of four specimens of uranium dioxide in adjacent positions of the Chalk River reactor, two specimens of $UO_{2.00}$ were radially cracked with no apparent grain growth, whereas two specimens of $UO_{2.15}$ were radially cracked and had extensive grain growth. In addition, the amount of fission gas released from the nonstoichiometric oxide was 100 to 200 times higher than that released from the near-stoichiometric specimens. Table VIII-1 is a compilation of the effect of oxygen to uranium ratio on fission-gas release as presented in the paper by Runnalls.[8]

Effect of Fabrication Method

It was also reported by Runnalls that some of the discrepancies in reported values of thermal conductivity may be due to the fabrication methods used in preparing stoichiometric UO_2 pellets. Table VIII-2 shows the variation in thermal conductivity at 140°F for UO_2 specimens prepared by different methods. It can be noted that the small change in the oxygen to uranium ratio (2.01 ± 0.01 to 2.03 ± 0.01) is sufficient to change the thermal conductivity of the specimen by about 25 per cent.

It is quite apparent from the variation of thermal-conductivity data that care must be taken in specifying the oxygen to uranium ratio in order to maintain the highest thermal conductivity and lowest fission-gas release possible. It would also appear that the effective thermal conductivity for UO_2 pellets under typical reactor operating heat fluxes and center temperatures of >2550°F is still near the previously reported value of about 1.0 ± 0.2 Btu/(hr)(ft)(°F).

Increased Thermal Conductivity of UO_2

Since an economic and performance gain can be obtained by increased thermal conductivity of UO_2, several mixtures of UO_2 and other

Table VIII-1 EFFECT OF OXYGEN TO URANIUM RATIO ON FISSION-GAS RELEASE FROM URANIUM OXIDE ELEMENTS CLAD IN ZIRCALOY-2 AND IRRADIATED IN TEST LOOP X-2-N[8]

Specimen identity	Oxide preparation method	Oxygen to uranium ratio	Average density, g/cm^3	Burn-up, Mwd per metric ton of U	$\int_{400°C}^{center} k(\theta)\, d\theta$, watts/cm	Fission-product xenon release, %
AG	Steam-sintered for 4 hr at 1325°C, then hydrogen sintered for 1 hr at 1325°C	2.002	10.41	4600	27.1	0.05
AK	Steam-sintered for 4 hr at 1325°C, then hydrogen sintered for 1 hr at 1325°C	2.002	10.45	4500	26.8	0.13
AA	Steam-sintered for 4 hr at 1325°C	2.15	10.47	4300	25.4	20.0
AS	Steam-sintered for 4 hr at 1325°C	2.15	10.48	4300	25.4	18.5

Table VIII-2 EFFECT OF SINTERING METHOD ON THE THERMAL CONDUCTIVITY OF URANIUM OXIDES[8]

Oxygen to uranium ratio	Density, g/cm^3	Method of fabrication	Thermal conductivity at 140°F,* Btu/(hr)(ft)(°F)
2.01 ± 0.01	10.3	Sintered in hydrogen at 3000°F for 2 hr; cooled in hydrogen	4.05 ± 0.29
2.03 ± 0.01	10.45	Sintered in steam at 2550°F for 3 hr; cooled in hydrogen	2.95 ± 0.17
2.01 ± 0.01	10.45	Sintered in steam at 2550°F for 3 hr; annealed in hydrogen at 2550°F for 2 hr; cooled in hydrogen	3.87 ± 0.12
2.21 ± 0.01	10.6	Sintered in steam at 2550°F for 2 hr; cooled in steam	0.98 ± 0.06

*Corrected to theoretical density.

oxides have been investigated in an attempt to obtain an increase in thermal conductivity without substantially decreasing the melting point or increasing the parasitic absorption of the fuel.

Several methods of increasing the thermal conductivity were studied by Sylcor,[13] under a contract from the AEC's Division of Reactor Development. Included in the study were the addition of high-thermal-conductivity metals and oxides.

Metallic Additions. Uranium dioxide dispersions in the metals molybdenum, niobium, and magnesium were studied by Sylcor. Several of the major properties of these materials are shown in Table VIII-3. Specimens of 67 vol.% UO_2 and 33 vol.% niobium and molybdenum were prepared. Densities of the test specimens never exceeded 87 per cent of theoretical density.

Table VIII-3 SOME PROPERTIES OF METALS CONSIDERED AS ADDITIONS[13] TO UO_2

Metal	Density, g/cm^3	Melting temp., °F	Thermal conductivity at room temp., Btu/(hr)(ft)(°F)	Thermal-neutron cross section, barns
Mo	10.2	4762 ± 90	83.6	2.4
Nb	8.57	4380 ± 27	38.7	1.1
Mg	1.74	1202 ± 3.6	92	0.063

Although addition of molybdenum to the UO_2 would improve its thermal conductivity, the resulting decrease in UO_2 volume per pellet and the high thermal-neutron cross section of molybdenum are undesirable. Niobium has a considerably lower thermal-neutron cross section, but it is questionable whether even this material would yield a more attractive fuel element because of the decrease in UO_2 volume and because of the relatively small gain in thermal conductivity.

Sylcor's attempt to infiltrate porous UO_2 with magnesium at high temperature resulted in only slight wetting of the UO_2 with no infiltration. In

Table VIII-4 BODY COMPOSITIONS[13]

	Batch Y-1	Batch N-1	Batch U-1	Batch M-1	Batch M-2	Batch C-1	Batch C-2
Additive, g:							
$UO_{2.28}$	259	334		298	347	347	
$UO_{2.36}$			350				250
Y_2O_3	2.26						
Nb_2O_5		3.32					
MoO_3				6.63	1.87		
CeO_2						2.24	1.62
Milling time, hr	6	6	0	6	6	6	6
Elvanol 51-05, g	1.2	1.5	1.5	1.5	1.5	1.5	1.2
Milling time, hr	2	2	0	2	2	2	2
Theoretical density,* g/cm^3	11.16	11.21	11.18*	11.18*	11.28	11.23	11.14
Sintered density, g/cm^3	8.44	9.20	10.99	6.02		9.57	9.23
Absorption, %	3.8		1.0	5.8		0.01	0.0
Sintered oxygen to uranium ratio		2.25	2.10	2.25		2.20	
Uranium, %		79.2	86.2	79.5		78.3	
Additive, %		0.39		0.4		0.55	
Sintering shrinkage, %:							
Diameter	15.6	17	18.3			17.2	
Length	16.6		17.7			16.8	

*Theoretical densities interpolated from data in J. J. Katz and E. Rabinowitch, "The Chemistry of Uranium, Part I. The Element, Its Binary and Related Compounds," p. 261, Division VIII, Volume 5, National Nuclear Energy Series, McGraw-Hill Book Company, Inc., New York, 1951; and in "Chemical Rubber Handbook," for density values of Y_2O_3, Nb_2O_5, MoO_3, and CeO_2.

Table VIII-5 THERMAL CONDUCTIVITY[13] OF NONSTOICHIOMETRIC UO_2*

Additive	Additive, in UO_2, at.%	Oxygen to uranium ratio	Density, g/cm^3	Per cent of theoretical density	Temperature of measurement, °C	Temperature of measurement, °F	Thermal-conductivity ratio, %†
Y^{+3}	2.08	2.2	8.44	76	516	961	150–155
Nb^{+5}	2.04	2.25	8.85	79	327	620	110–115
Nb^{+5}	2.04	2.25	8.85	79	528	982	150–160
None		2.00	10.62	97	360	680	125
None		2.00	10.62	97	422	792	133

*All specimens compared with $UO_{2.10}$ standard samples (density 10.99 g/cm^3, 98.5 per cent of theoretical density).
†(Thermal conductivity of sample × 100)/(thermal conductivity of $UO_{2.10}$).

addition, the low melting temperature of magnesium would prevent its use in very high fuel temperature applications.

Oxide Additions to UO_2. A second Sylcor study investigated the possibility of increasing the thermal conductivity of UO_2 by changing its semiconductor properties by the addition of higher or lower valence materials of low thermal-neutron cross section. Several different compositions of materials which would form extensive solid solutions with UO_2 were prepared. Specimens consisted of $UO_{2.28}$ or $UO_{2.36}$ and the following oxides: Y_2O_3, Nb_2O_5, MoO_3, and CeO_2. Some of the specimens fabricated by Sylcor for test purposes, and their compositions, are listed in Table VIII-4.

Thermal-conductivity measurements on some of these specimens were made by a modified Francl-Kingery method described in reference 13. The results of these measurements are shown in Table VIII-5.

As reported by Sylcor, the tests made with samples Y-1, containing less than 1 wt.% Y_2O_3, and N-1, containing 1 wt.% Nb_2O_5 (Table VIII-4), showed that the thermal conductivity measured at approximately 955 to 980°F was 50 to 60 per cent higher than that of the standard specimen. The standard specimen had the composition $UO_{2.10}$ and a bulk density of 10.99 g/cm^3 (98.5

per cent of theoretical). The oxygen content of the standard should, of course, be noted. At approximately 791°F the conductivity of stoichiometric UO_2 is about 33 per cent higher than that of the standard. Compared to stoichiometric UO_2, the conductivity of the $UO_{2.28}$-Y_2O_3 specimens is higher by only 13 to 16 per cent, and that of the $UO_{2.28}$-Nb_2O_5 is higher by only 16 to 20 per cent. In addition, the specimens tested had densities of 8.44 and 9.2 g/cm^3 as compared to the $UO_{2.00}$ density of 10.62. These changes in density decrease the uranium content of the fuel pellet, relative to that of $UO_{2.00}$, by approximately 20 per cent for the UO_2-Y_2O_3 specimens and approximately 13 per cent for the Nb_2O_5 specimen. The decrease in uranium content partly cancels the gain from increased thermal conductivity and is undesirable for other reasons. Consequently it cannot be said that the specimens tested represent obvious improvements over pure stoichiometric UO_2 elements.

After further experimentation, Sylcor fabricated high-density specimens of essentially stoichiometric UO_2 with various additives.[16] A list of the various specimens fabricated is shown in Table VIII-6.

Table VIII-6 SAMPLES FOR THERMAL-CONDUCTIVITY TESTS[16]

Sample composition, mole %	Density, g/cm^3	Per cent of theoretical density	Oxygen to uranium ratio
UO_2 + 1/4 Y_2O_3	10.53	96.5	
UO_2 + 1 Y_2O_3	10.42	96.4	2.00
UO_2 + 4 Y_2O_3	9.67	91.6	
UO_2 + 1/4 Nb_2O_5	10.34	94.6	2.00
UO_2 + 1 Nb_2O_5	10.47	96.5	
UO_2 + 1/4 CeO_2	10.55	96.5	2.00
UO_2 + 1 CeO_2	10.23	94.0	

Relative thermal-conductivity data obtained on samples of UO_2 + 4 mole % Y_2O_3 and UO_2 + 1 mole % Y_2O_3 as compared to stoichiometric UO_2 are shown in Table VIII-7.

These data indicate that the addition of 4 mole % Y_2O_3 to UO_2 increases the thermal conductivity of UO_2 about 32 to 33 per cent at 1582°F; the addition of 1 mole % Y_2O_3 to UO_2 shows an increase of between 26 and 41 per cent at 2228°F but a decrease of the order of 10 per cent at 543°F.

Although the data obtained are quite encouraging and do indicate that the thermal conductivity of UO_2 is increased by the addition of Y_2O_3, more refined data and further experiments are required to substantiate the conclusion. Insufficient data are available as to the effect of the additive on the melting temperature of the samples. A decrease in the melting temperature of the sample combined with the slightly lower uranium content of the sample as compared to stoichiometric UO_2 could offset the gain in thermal conductivity.

Table VIII-7 RELATIVE THERMAL CONDUCTIVITY[16] OF UO_2 PLUS Y_2O_3

Sample composition, mole %	Temperature °C	Temperature °F	Relative thermal conductivity compared to UO_2
UO_2 + 4 Y_2O_3	861	1582	1.32–1.33
UO_2 + 4 Y_2O_3	826	1529	1.26–1.31
UO_2 + 4 Y_2O_3	740	1364	1.31–1.33
UO_2 + 4 Y_2O_3	656	1213	1.05–1.25
UO_2 + 1 Y_2O_3	1220	2228	1.26–1.41
UO_2 + 1 Y_2O_3	571	1062	0.93–1.00
UO_2 + 1 Y_2O_3	406	763	0.94–0.95
UO_2 + 1 Y_2O_3	284	543	0.84–0.91

To obtain data which will establish that the addition of Y_2O_3 to UO_2 improves the thermal performance, samples of stoichiometric UO_2 with and without Y_2O_3 additives have been prepared by Sylcor for insertion into the Hydraulic Rabbit loop at Chalk River.[17] Samples will be prepared of such enrichment that the central temperatures will cause grain growth in the UO_2 sample and no grain growth in the UO_2-Y_2O_3 sample provided the UO_2-Y_2O_3 sample has both a higher thermal conductivity than UO_2 and approximately the same melting temperature as UO_2.

Steam Corrosion Data for UO_2

To obtain some data on the effect of steam corrosion on the samples of UO_2 with oxide additives, Sylcor carried out corrosion tests with steam at 636°F and 2000 psi.[18] The results of these tests, shown in Table VIII-8, indicate that no significant weight changes occur.

In water reactors, where the hot UO_2 fuel will be exposed to steam if a cladding defect occurs, the corrosion behavior of the UO_2 in steam is important. Runnalls[8] reported that WAPD analyzed a sample of initially stoichiometric UO_2 from an irradiated defected fuel rod and found

Table VIII-8 STEAM CORROSION OF UO_2 PLUS ADDITIVES*[18]

Sample composition, mole %	Per cent of theoretical density	Oxygen to uranium ratio	Weight change, mg/(cm²)(48 hr)†
UO_2	97	2.00	+0.1
UO_2 + 1/4 CeO_2	96	1.99	+0.1
UO_2 + 1 CeO_2	96		−0.2
UO_2 + 1/4 Y_2O_3	96		−0.1
UO_2 + 1 Y_2O_3	94	2.00	0
UO_2 + 1/4 Nb_2O_5	95	2.00	−0.6
UO_2 + 1 Nb_2O_5	94	2.00	−0.2

*Carried out at 636°F, 2000 psi, for 48 hr.
†±0.1 mg/(cm²)(48 hr).

that the average postirradiation composition was $UO_{2.06}$.

Confirming data from Chalk River was also presented by Runnalls. A sample of UO_2 from the cooler region of a defected fuel rod underwent a composition change from $UO_{2.00}$ to $UO_{2.21}$ during irradiation in a pressurized-water loop.

Since the thermal conductivity appears to decrease with increased oxygen to uranium ratio, a fuel-element cladding failure apparently may lead to central melting of the UO_2 if the center temperature was near the melting temperature before the failure.

Oxidation of UO_2 in CO_2

Measurements of the oxidation rate of UO_2 in CO_2 by Antell et al.[19] were also presented by Runnalls.[8] The oxidation of UO_2 by CO_2 at 1292°F caused a weight gain of sintered UO_2 pellets (density 9.6 g/cm³) of 0.008 mg/(cm²)(hr). For comparison, the corrosion of uranium metal by CO_2 under similar conditions caused a weight gain of 400 to 560 mg/(cm²)(hr).

Bibliography of UO_2 Data

For more details and an extensive bibliography of reports on UO_2, the reader is referred to references 5, 8, 13, and 20 to 24.

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PROGRESS ON SPECIFIC REACTOR TYPES

HEAVY-WATER REACTORS

Closely related to the subject of breeders (discussed elsewhere in this issue) is that of reactors having high conversion ratios. With recycling of converted fissionable isotope, the reactor with high conversion ratio can extend by a large factor the energy yield from each ton of nuclear fuel mined. If an effective conversion ratio, $\bar{R}_c$, is defined as the number of fissionable atoms produced per fissionable atom destroyed, taking into account all losses of fissionable isotope in diffusion plants, chemical plants, and fuel-processing plants, the amount of energy produced per ton of natural uranium is approximately

$$E = \frac{5000}{1 - \bar{R}_c} \text{ Mwd/ton}$$

As $\bar{R}_c$ approaches unity, the availability of fissionable isotope is no longer the factor which limits energy production, but the supply of fertile isotope becomes limiting, and the energy production per ton of natural uranium approaches a limit of something like 700,000 Mwd/ton. Thus the energy production per ton of natural fuel attainable with a converter reactor might approach that attainable with a breeder if nonreactor losses of fissionable isotope were sufficiently low. The practical distinction between the breeder and the converter is that the breeder can produce excess fissionable isotope, for use as inventory in expanding the total nuclear-power-generating capacity, or for use in reactors of lower neutron economy.

Although the attainment of breeding in thermal reactors is quite difficult, particularly in solid-fuel thermal reactors, the attainment of conversion ratios sufficiently high to extend substantially the energy yield per ton of natural uranium is considerably easier. D_2O is particularly favorable as moderator for high-conversion-ratio reactors, and it was in this area that most of the interest centered during the symposium on D_2O-moderated power reactors[1] (Mar. 3 and 4, 1959), although the D_2O homogeneous breeder was also discussed.[2]

The D_2O-moderated reactor can also be designed to operate with natural-uranium fuel at a relatively high conversion ratio, although the maximum conversion ratio will no doubt be attained when some enrichment of the fuel is used. There are a number of D_2O-moderated reactor types which can be designed either for natural-uranium operation or, alternatively, for operation with enriched uranium at higher conversion ratio. Thus the development of the D_2O technology will make these two objectives attainable, and the choice between the two can be made for individual reactor designs as the economic and fuel availability considerations of the moment may dictate. The commercial incentives for development of natural- or near-natural-uranium reactors were given by W. L. Webb, in connection with his discussion of the East Central Nuclear Group–Florida West Coast Nuclear Group gas-cooled, D_2O-moderated reactor:[3]

1. Basically the cost of U^{235} in natural uranium or very slightly enriched uranium is lower than its cost in more highly enriched uranium. Thus the possibility of very low fuel cost exists, either with natural uranium alone or with natural uranium plus plutonium recycle.

2. The probability that natural uranium will become a free market commodity is greater than the probability that enriched uranium will be available in a free market. Thus it is less probable that the user of natural uranium will have the cost of his fuel burn-up or the value of his fuel inventory fixed by government regulations.

3. Chemical processing of the fuel and sale of plutonium are not necessary to achieve low fuel

cost with natural- or near-natural-uranium reactors. Thus the operator is not subject to the uncertainties of chemical-processing costs and plutonium buy-back prices.

Other incentives, which relate to the broader aspects of fissionable-isotope availability, were discussed by Menke.[4]

1. We should not build into our power economy a potential conflict between a military demand for enriched material and a utility demand for the same material.

2. Reactors which operate on natural uranium are capable of producing plutonium in large quantities with great facility under any emergency demand. Thus natural-uranium reactors are emergency sources of plutonium production which are independent of isotope-separation plants and which can most economically be operated at large throughput to maximize production.

3. These emergency sources of plutonium production, if they are part of the power economy, are dispersed, rather than large single-target, production centers.

4. This plutonium-production capability can be made available without a large capital expenditure beyond that which would normally go into the nuclear-power capability.

In most D_2O reactors the inventory of D_2O constitutes a significant capital investment. The price of D_2O is fixed at $28 per pound, and this price along with the specific power of the reactor determines the cost of the D_2O in inventory. The annual rate of charge which must be made against the value of the inventory will depend upon how the operator of the reactor is able to finance the cost, upon the profit which he must make to be consistent with his financing system, and upon the taxes which may be applicable in his particular location and circumstance. For evaluation studies, it is necessary to use some charge rate which is, hopefully, representative. In recent studies the rate has been 12.5 per cent per year. At the D_2O conference, Edward Shepherd,[5] of the AEC Division of Finance, summarized how this charge rate was arrived at.

A 6 per cent annual charge is attributable to the return on invested capital. This rate was arrived at as an average, based on Federal Power Commission (FPC) statistics, of the financing pattern of individual utilities. The financing pattern is as follows: 50 per cent in long-term bonds with a yield of 4 per cent per annum, 15 per cent in preferred stock with an average dividend of 4.6 per cent, and 35 per cent in common stock with earnings applicable to common shareholders of 9.4 per cent.

An annual charge of 6.04 per cent is estimated to result from taxes. This estimate, again based on FPC statistics, comprises 3.61 per cent for Federal income taxes, 0.1 per cent for other Federal taxes, and 2.33 per cent for state and local taxes, 65 per cent of which are ad valorem or property taxes.

An annual charge of 0.52 per cent results from insurance costs. Of this, 0.25 per cent is the average expenditure in conventional plants for insurance for fire, extended coverage, boiler, vandalism, etc. The remaining 0.27 per cent is property insurance on the plant itself (in this case the heavy water) for protection against losses of plant and personnel resulting from nuclear hazards. No charges are made for third-party liability insurance, for the obvious reason that the heavy water makes no additional contribution to the third-party hazards.

The above charges add up to a total of 12.56 per cent per annum, which is usually rounded off to 12.5 per cent and which is applied against the inventory of D_2O. It does not include make-up costs or costs associated with maintaining the quality of the D_2O. It does not include any charge for depreciation, since the D_2O does not depreciate in effectiveness if its quality is maintained and since there is no basis on which to predict a change in the unit price of D_2O.

The question of how large a power-production capacity in D_2O reactors could be supported by the present D_2O production plants was discussed at the conference.[6] There are two plants in the United States having approximately equal capacity and a combined capacity of approximately 1000 tons of D_2O per year. One of these, the Dana plant, is not in operation but is held in stand-by condition. The other, at Savannah River, is said to be operating at reduced capacity. It appears that a conservative estimate of the D_2O inventory required per electrical megawatt of power-production capacity in solid-fuel reactors would be approximately 1 ton per electrical megawatt. Thus, if both production plants were operating at full capacity, their production would initially support the construction of about 1000 Mw(e) of power capacity per year. This would represent an annual addition of electrical capacity equal to about 1 per cent of the

present electrical capacity of utilities in the United States (at 80 per cent plant factor). As the D_2O reactors operated, some of the D_2O production capacity would be needed to supply make-up losses. If these are assumed to amount to 2 per cent of the total inventory per year, the entire output of one production plant would be required for make-up to the D_2O reactors after enough reactors had been built to supply about one-tenth the present electrical consumption of the United States. Thus the existing production plants, although they could not supply a D_2O electric generating system comparable to the present utility system in the United States, could nevertheless supply sufficient D_2O to support a very substantial power-generation capacity in D_2O-moderated reactors.

The operating experience of the Du Pont Company with the Savannah River reactors was discussed by Babcock,[7] with particular attention to the losses and cleanup problems of D_2O. He stated that the annual losses amount to about 3 per cent of the total heavy water in the system per year and that this loss rate, when averaged over a long period of time, is very nearly the same for all of the five reactors at the SRP. The losses are approximately equally divided between a few large ones (e.g., sudden large leaks which may account for several hundred gallons before they can be detected and stopped) and many small ones. Of the latter, the D_2O carried out with the fuel elements during discharge is a major fraction and amounts to perhaps one-third of the total losses. The expenditure of D_2O in deuterating the resins in the ion exchangers does not amount to an important contribution to the D_2O losses.

The inleakage of H_2O into the D_2O amounts to 5 or 6 lb per day per reactor. This water is removed in a still,[8] where it emerges as a 25 per cent H_2O–75 per cent D_2O mixture. This mixture is then returned to the main D_2O production plant for enrichment of the D_2O up to its normal usable concentration. The cost of this cleanup of H_2O from the D_2O system is much less than the cost resulting from direct D_2O losses.

In its work toward the development of D_2O power reactors, the effort of the Savannah River group[7] is unique in that a strong effort is being made to develop metallic fuel elements. It is anticipated that the metallic fuel elements, which are to be fabricated in long sections by coextrusion with a zirconium or Zircaloy cladding, will be considerably cheaper to fabricate than oxide elements. The other incentive for the use of metal, of course, is the higher reactivity attainable with uranium metal fuel than with uranium oxide. It is hoped that the metallic fuel elements can be made in quantity for as little as $7 per pound of uranium, including the cost of the zirconium in the cladding. By quantity production is meant something like that which would supply about 10 power reactors of 100-Mw(e) size. If fuel-fabrication costs approaching this low figure can be achieved, then the fuel cost with natural uranium can be quite low despite the relatively short fuel lifetime that can be expected with metallic fuel elements.

Several types of D_2O-moderated reactors were discussed at the conference, including: The sodium-cooled reactor,[9] the Carolinas-Virginia D_2O-cooled reactor,[10] the gas-cooled reactor of the East Central Nuclear Group–Florida West Coast Nuclear Group,[3] the aqueous homogeneous reactor,[2] the organic-cooled reactor,[11] the boiling D_2O reactor,[12] and the Sargent & Lundy study[13] of several D_2O reactor types. The latter study was reviewed in *Power Reactor Technology*, 2(4): (September 1959).

Heavy-water Components Test Reactor

The Heavy-water Components Test Reactor (HWCTR)[14–17] is a high-temperature, high-pressure, pressure-vessel reactor, cooled and moderated by D_2O, whose purpose is to test fuel elements and other components for large D_2O systems. It can accommodate 12 test fuel elements of natural or slightly enriched uranium, about 10 ft long, in a region near its center. This region is surrounded by a ring of enriched-fuel "driver" elements to furnish sufficient reactivity for operation. The pressure vessel, 7 ft 2 in. in diameter in the reactor region, is designed for 315°C and 1500 psi; the maximum exit operating temperature of D_2O coolant from the pressure vessel is 285°C.

By the end of December 1959, approximately 19 per cent of the construction and 84 per cent of the firm design of the HWCTR was complete. A recent picture of the containment for the reactor can be found in reference 18. The start-up goal is early in 1961, and the total cost of the reactor complex, including two in-pile coolant loops, is estimated to be about $9,300,000.

The in-pile coolant loops were originally scheduled for installation some time after the start-up of the reactor; the AEC, however, recently requested that two of these loops be installed at the time of constructing the HWCTR. It is planned that one loop will be cooled with liquid D_2O and the other with boiling D_2O, with the option that the latter loop be convertible, at some future date, to operation by D_2O steam or CO_2. The loops will have their own make-up, purge purification, and heat-removal systems independent of the reactor systems. Stainless steel will be used for the piping and equipment in the D_2O loops. That part of the loop within the pressure vessel proper consists essentially of two coaxial tubes, i.e., a "bayonet," so arranged that the coolant enters at the top, flows down the annulus to the bottom, reverses, and passes up through the test fuel element and out of the pressure vessel.* The portion of the bayonet that surrounds the test fuel assembly within the core is of Zircaloy-2 with stainless-steel end fittings. The over-all length of the bayonet is about 26 ft, and the active core length available for a test element is about 9 ft 5 in. The boiling loop will operate at nominal reactor pressure (about 1000 psi), and the liquid loop will operate about 500 psi above reactor pressure. The steam-D_2O mixture from the boiling loop is passed out of the reactor vessel and into a small quench chamber where cold D_2O from a heat exchanger in a bypass loop is mixed with the steam-D_2O mixture to condense the steam. The liquid coolant then passes into a surge tank and then back to the pumps. The fuel element in the boiling loop can generate up to about 1.4 Mw, whereas the element in the liquid loop can generate about 1.8 Mw.

The prototype "driver" fuel tube for the HWCTR has been fabricated. The fuel tube has a meat composed of zirconium–9.3 wt.% uranium (93 per cent enriched) clad inside and out with Zircaloy. The tubes were formed by extrusion; some scoring of the cladding by the die was observed in two of the extrusions, but enough satisfactory tubes were manufactured to undergo irradiation testing.

*The actual details of the bayonets provide for, in addition to the inner and outer bayonet tubes, an insulating liner, a shield sleeve, and a hold-down tube. One of the loop bayonets will be of Zircaloy-4; the other of Zircaloy-2.

The Sodium Deuterium Reactor

On Feb. 5, 1959, the Sodium Heavy-water Reactor Task Force was formed to review the status and potential of the natural uranium–sodium–heavy water reactor concept. Reference 19 is the report of that Task Force. Briefly, the purpose of the Task Force was to evaluate the reactor concept in relation to other reactors employing natural uranium as the fuel material. Large power reactors, 200 Mw(e), were critically reviewed to determine the immediate and long-range potential of the concept.

The basic reactor design studied was a pressure tube–calandria type. Heavy water is contained in an aluminum vessel, the calandria, which is penetrated by a number of aluminum tubes. Coaxial with these aluminum calandria tubes are the pressure tubes containing the fuel elements. Between the pressure and calandria tubes is located a barrier tube, which separates the coolant and moderator in the event of both a fuel- and calandria-tube failure. The barrier tubes also act as a thermal radiation shield. Through the fuel tube is pumped liquid sodium at a pressure of about 165 psig. Nitrogen is flushed between the fuel and barrier tubes and the calandria and barrier tubes; the streams are monitored to detect any leakage of coolant and/or moderator. The pressure tubes are arranged in a vertical geometry with crossover piping connecting each fuel-coolant tube with a coolant header both above and below the core. A sodium-to-sodium heat exchanger is employed as an intermediate, with a sodium-to-water exchanger employed for steam generation.

Two reference designs of sodium D_2O reactors were chosen for study. These are as follows:

1. A large-scale plant requiring the least amount of research and development before construction that would be operable in the middle or late 1960's.
2. An advanced version that could be operable in the early 1970's.

Both designs were chosen to operate with natural-uranium fuel and to utilize effectively the high-temperature capabilities of the sodium coolant. Details on the two designs, designated A and B, are found in Table IX-1. Both are single-region machines. The basic fuel element for reactor A is a rod; a cluster of over 100 rods constitutes a fuel bundle. The right-circular

Table IX-1 DETAILS OF SODIUM D_2O REACTORS [200 MW(E)]

Parameter	Reactor A	Reactor B
Fuel	2 wt.% Zr-U alloy	UC
Fuel cladding	Zircaloy-2	Nb
Fuel tube	S.S.	Nb
Barrier	2-S Al	Zircaloy-2
Sodium outlet temperature, °F	800	1150
Thermal efficiency, %	30	37
Fuel inventory, metric tons	117	95
D_2O inventory, metric tons	150	115

cylindrical core has a height of about 20 ft and a diameter of about 18 ft; the resulting fuel rod is 10 ft in length. The individual rods are separated by helical wire spacers wound around the rods. The rods are maintained in a hexagonal array in the bundle by a top and a bottom spacer, and two clusters are vertically supported inside each fuel tube from the bottom.

One of the major problem areas of the concept is that of attaining sufficient reactivity for operation with natural uranium. This problem, common to all natural-uranium reactors, is intensified by the use of sodium and by the desire to utilize the high coolant temperatures which are potentially possible with sodium. The sodium itself absorbs a significant fraction of the available neutrons, and the materials of the fuel tubes, barrier tubes, and calandria tubes account for further important absorption. These neutron losses appear to be so high that a high-density fuel which can be used in rather large slugs is necessary for the achievement of sufficient reactivity. Such a requirement apparently eliminates UO_2 as a fuel material, and a proven high-density fuel is not presently available. The necessity for minimizing absorption also accentuates the problems of supporting the fuel bundles and the fuel tubes. The desire to minimize these absorption problems leads to the rather large fuel clusters used in the designs studied, but the gains which can be achieved in this way are limited by the self-shielding of the fuel and the resulting undesirable power distribution within the fuel cluster.

The unique problem of the concept is, of course, the development of methods of preventing hazardous contact between the sodium and the liquid D_2O. Considerable progress has been made on this problem in the course of the sodium D_2O project. The problem, like many of the problems related to reactor hazards, requires proof of safety under all reasonably conceivable circumstances, and it is therefore particularly difficult.

Further problems of the concept are those common to sodium systems, chief of which perhaps is the steam generator.

The estimated capital costs for the two reactors studied were $97,167,000 for reactor A and $80,213,000 for reactor B. These correspond to $453 per kilowatt (electrical) and $375 per kilowatt (electrical), respectively. The structural costs include some six million dollars for a reactor containment building, a concrete foundation slab, and the associated concrete piles and walls. About one-half million dollars, in addition, is budgeted for construction of a shadow shield since the Task Force assumed the plant was to be built in a populated area. The sodium-water heat exchangers in reactor A are of the double-wall type, which cost about $150 per square foot of heat-transfer area. The steam generators for reactor B are assumed to be of single-wall design costing $50 per square foot.

The estimated fuel costs and power-generation costs are shown in Table IX-2. Because of the uncertainty in calculating exposure limits, the table contains estimates as to the "reasonable" maximum and minimum fuel costs derived by the Task Force.

The conclusions given by the Task Force are as follows:[19]

1. The potential of a sodium heavy-water reactor concept for the economical production of electric power in the near future does not appear promising. A return of 30 per cent efficiency, which is claimed for reference reactor A, is not sufficient gain to overcome the risk of marginal reactivity, unreliable or expensive steam generators, and lack of confidence in the fuel assembly.

2. Other natural-uranium reactors, such as the pressurized- or boiling heavy-water concepts, have reached a stage of development higher than that of the sodium-cooled concept. The expenditure of research and development funds to solve the previously reported problems solely for the sodium heavy-water reactor and to achieve a full-scale operating plant would be greater for this concept than for other natural-uranium power reactors.

3. Sodium, when considered as a separate entity, has the potential of producing high-quality steam, which is synonymous with high thermal efficiency. However, sodium is not unique in having this characteristic. Other natural-uranium systems, such as those employing nuclear superheat, have the poten-

Table IX-2 ESTIMATED FUEL COSTS AND POWER-GENERATION COSTS[19] FOR 200 MW(E) SODIUM DEUTERIUM REACTOR

	Reference reactor A				Reference reactor B			
	High burn-up, 3300 Mwd/ton		Low burn-up, 2000 Mwd/ton		High burn-up, 4800 Mwd/ton		Low burn-up, 2000 Mwd/ton	
	Minimum	Maximum	Minimum	Maximum	Minimum	Maximum	Minimum	Maximum
Fuel cost:								
Replacement cost, thousands of dollars per year	6,450	7,800	10,600	12,900	3,130	6,350	7,520	15,300
Fissionable-material working capital, thousands of dollars per year	468	468	468	468	382	382	382	382
Nonfissionable-material and fabrication working capital, thousands of dollars per year	545	812	545	812	359	1,180	359	1,180
Net Pu credit, thousands of dollars per year	(432)	(432)	(268)	(268)	(442)	(442)	(186)	(186)
Total, thousands of dollars per year	*7,031*	*8,648*	*11,345*	*13,912*	*3,429*	*7,470*	*8,075*	*16,676*
Fuel cost, mills/kw-hr	5.00	6.20	8.10	9.90	2.40	5.30	5.80	11.90
Fixed charges on investment, mills/kw-hr	9.25	9.25	9.25	9.25	7.68	7.68	7.68	7.68
Fixed charges on heavy water, mills/kw-hr	0.94	0.94	0.94	0.94	0.71	0.71	0.71	0.71
Heavy-water losses, mills/kw-hr	0.04	0.04	0.04	0.04	0.03	0.03	0.03	0.03
Operation and maintenance, mills/kw-hr	1.20	1.20	1.20	1.20	1.20	1.20	1.20	1.20
Total generation cost, mills/kw-hr	*16.43*	*17.63*	*19.53*	*21.33*	*12.02*	*14.92*	*15.42*	*21.52*

tial of producing high-quality steam. Also, these other systems do not have the disadvantages incurred with the use of sodium, namely, marginal reactivity and costly steam generators. In addition, the inherent characteristics of the concept do not lead to power costs that are lower than those of other natural-uranium heavy-water reactors. Hence it does not appear that the sodium heavy-water reactor concept has a long-range potential for the economical production of electric power that is better than other natural-uranium power reactors.

4. Current and projected research and development programs for other reactor systems may produce results in the future which can be applied to the sodium heavy-water reactor concept. If this occurs, the concept may appear more favorable when it is compared with other natural-uranium reactors.

5. The conclusions drawn by the Sodium Heavy-water Reactor Task Force are strictly valid only at the time of this evaluation and in the immediate future. When major advancements occur in sodium and natural-uranium heavy-water technologies leading to lower power costs, many areas of reactor technology can be affected. The operation of sodium plants under construction may result in significant cost reductions that can be applied to the sodium heavy-water reactor concept or to any other sodium-cooled concept. Natural-uranium fuel-element development may lead to similar cost reductions. If these occur, a reevaluation of the concept should be made at that time.

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Fuel Research

Several reports have been published recently presenting the results of development programs for sodium graphite reactor fuel elements. Reference 1 reports on capsule irradiation experiments done in the MTR. Cylinders $\frac{3}{8}$ in. in diameter by $1\frac{3}{8}$ to $1\frac{1}{2}$ in. in length were placed stainless-steel capsules containing sufficient NaK to cover the specimens. Concentric with the specimens were two thermal shields to assist in raising the NaK temperatures to desired values. The specimen was centered and supported on a thermocouple thimble. It was desired to attain central metal temperatures of about 1200°F with a surface temperature of 950°F. The principal results of the irradiations were dimensional changes of the fuel cylinder and corresponding density changes.

The materials used for the study are as follows: powder compacts of uranium–1.2 wt.% molybdenum; cast uranium–2 wt.% zirconium; alpha-rolled, beta heat-treated unalloyed uranium; cast and machined, or cast and swaged, thorium–11 wt.% uranium.

An unalloyed uranium and a uranium-zirconium alloy specimen were irradiated to a burn-up of about 0.3 total at.% with an average centerline temperature of 950°F. The specimens could not be removed from their capsules owing to swelling and were discarded. The uranium-molybdenum alloy specimens were irradiated to about 0.3 total at.% burn-up at a centerline temperature of 1200°F and exhibited volume increases up to 31 per cent. The thorium-uranium alloys were irradiated to about 1 per cent burn-up at a centerline temperature of about 1200°F and exhibited a total volume increase of 6 per cent. Based on these experiments the authors conclude that: "Of the materials tested, only the thorium–11 wt.% uranium fuel has shown promise as a high-temperature fuel material."

Reference 2 presents information obtained from operation of the Sodium Reactor Experiment (SRE). The operating conditions for the SRE are given in Table X-1. At the conditions shown in Table X-1, the maximum fuel-slug centerline temperature is about 1000°F with a maximum slug surface temperature of about 850°F. An irradiation program was established to evaluate, by destructive and nondestructive testing, the behavior of the fuel elements of the first SRE core. The nondestructive testing consisted of the following hot-cell operations: straightness determination, snap gauging for cladding-diameter measurement, and visual observation. The elements were also viewed by remote television while still in the reactor vessel by raising the elements out of their channels. The destructive examinations consisted of the following:[2]

1. Disassembly and decanning, including fission-product analysis in the gas-filled expansion void and in the NaK bond.
2. Visual and photographic examination.
3. Dimensional measurement for length, diameter, and warp.

Table X-1 OPERATING CONDITIONS FOR THE SRE[2]

Type:	Thermal heterogeneous
Capacity:	20 Mw(t)
Moderator:	Graphite
Coolant:	Sodium T_{in} = 525°F (actual) T_{out} = 850°F (actual) Flow = 800 to 1200 gal/min
Fuel loading:	Core I 2.78 per cent enriched (U^{235}) unalloyed uranium Weight, 3000 kg
Fuel element:	No. of elements, 43 Rods per element, 7 Over-all length, 8 ft 6 in. Active length, 6 ft
Flux:	Core I Maximum thermal-neutron flux density, 1.5×10^{13} nv Average thermal-neutron flux density, 1.0×10^{13} nv Cell spacing 11 in. on triangular lattice array

4. Surface measurement.
5. Density determination.
6. Burn-up analysis by gamma-radiation detection techniques and radiochemical analysis.
7. Metallographic examination.

The standard SRE fuel rod consists of a number of 0.75-in.-diameter by 6.0-in.-long slugs of unalloyed uranium placed in a type 304 stainless-steel tube. The preexposure clearance between the slug and the tube is 0.010 in., and the jacket thickness is 0.010 in. The rods are wrapped with 0.092-in. wire to provide for circulation control and spacing. Included in the normal loading were several "defected" uranium slugs; these slugs had been fabricated to deliberately contain end cracks and heavy seams.

The results of the examinations are summarized as follows:[2]

> Nondestructive examinations of the unalloyed uranium fuel in the Sodium Reactor Experiment (SRE) showed no increase in cladding diameter after 850 Mwd/metric ton calculated maximum burn-up. Some clad stretching is evident at 950 Mwd/metric ton. Destructive examinations at maximum burn-ups of 65,105, 390, and 850 Mwd/metric ton showed progressive irradiation-induced dimensional changes. Data showed maximum changes in diameter of 1.7 per cent, in length of 2.9 per cent, and in density of −2.6 per cent for the highest burn-up. Surface conditions indicated progressive roughening with increasing fuel-material burn-up level. All surface and dimensional characteristics appeared to be directly related to burn-up. Some relationship was found between surface roughening and fuel-element surface temperature. The SRE fuel-element design appears to be performing well. Fuel-element cladding examinations indicated no apparent change in ductility after exposure to surface temperatures of about 850°F, with subsequent cycling resulting from start-up and shutdown of the reactor.

The author's conclusions are:[2]

OBSERVATIONS ON THE FUEL ELEMENT AS A WHOLE

1. The SRE fuel elements appear to be operating satisfactorily, as far as structural features are concerned. The limiting factor at this time appears to be the behavior of the fuel material.
2. Slug dimensional increases and density decreases are filling the 0.010-in. NaK-bond annulus and could ultimately rupture or cause the element to stick in the fuel channel.

OBSERVATIONS WITH RESPECT TO THE FUEL MATERIAL

1. The fuel has undergone dimensional changes (length and diameter) and has decreased in density. These changes are a maximum at the point of highest burn-up. These changes have not as yet endangered the element at 850 Mwd/metric ton maximum burn-up.
2. The presence of the fuel-slug fabrication defects (heavy seams and end cracking) did not appear to impair fuel-slug performance under irradiation.

Previous data on the irradiation behavior of uranium-molybdenum, uranium-zirconium, and thorium-uranium fuel elements which have been reviewed in previous issues of *Power Reactor Technology* (Vol. 1, Nos. 1 to 4) are in essential agreement with the above results. The predominant radiation damage which occurs in unalloyed fuel elements and alloy elements of high uranium content is a swelling type damage presumably caused by fission-product gases. The data on swelling type damage which were reviewed in *Power Reactor Technology*, 1(4): 32 (September 1958) show that a large amount of swelling per unit of burn-up is to be expected at high metallic uranium operating temperatures.

For more detailed data on various fuel and core materials, the reader is referred to the Technical Progress Review, *Reactor Core Materials*, prepared by Battelle Memorial Institute.

SRE Fuel-element Damage

A brief summary was given in *Power Reactor Technology*, 3(1): (December 1959) of the difficulties experienced in the SRE as a result of a Tetralin leak, which caused severe damage to some of the fuel elements. Since then the interim report of an Atomic International *Ad Hoc* Committee for investigation of the difficulties has been issued.[3]

The report of the Committee supports the conclusion that the difficulties were not due directly to any inherent characteristic of the sodium graphite reactor, but to the specific circumstance that Tetralin, an organic coolant, was used in an application where a relatively minor equipment failure allowed its leakage into the primary sodium system. The most general conclusion that can be drawn is that the

use of Tetralin or other organic is to be avoided where leakage into a sodium coolant system could occur. However, the report contains other interesting information. Since the significance of some of the information has not yet been determined, an adequate summary is difficult. Those engaged in the design or development of sodium-cooled reactor systems will find a detailed study of the report worthwhile.

The report contains brief summaries of the design characteristics of the reactor (reference 4 contains a more detailed description), of its operating history since start-up, and of the information gained from operation, as well as a detailed account of the abnormal operation and its results.

From its date of criticality, Apr. 25, 1957, through the end of June 1959, the SRE had logged a total of 2409 Mwd(t), equivalent to about 120 days of full-power operation. The last 260 Mwd was accumulated with the reactor outlet temperature at a level of 1000°F or higher, generating steam at 900°F and 600 psig. For a short period the reactor was operated (at reduced power) at an outlet temperature of 1065°F, generating steam at 1000°F. It is stated that the SRE proved easy to operate once the minor difficulties with the new sodium systems were overcome and that the reactor has exceeded its design performance and has successfully demonstrated the capabilities of the sodium graphite reactor concept.

It appears most probable that the damage to the fuel elements occurred between July 12 and July 16, 1959. In the report the chronology of operation up to this time is covered from Nov. 29, 1958, because conditions existed at that time which were in some respects similar to those during the July 12 to July 16 period.

Just prior to the start-up on Nov. 29, there had been a shutdown of about two months' duration for repairs and modifications. During these operations a substantial quantity of oxygen had found its way into the primary sodium. After start-up there was an excessive spread in the outlet temperatures of the various coolant channels. The spread between the maximum and minimum outlet channel temperatures amounted to as much as 385°F when the reactor was operating at only 3.6 Mw(t); the normal spread is less than 100°F at 20 Mw(t). The abnormal spread was attributed to plugging or partial plugging of coolant channels by sodium oxide. A high oxide content of the sodium was confirmed by the plugging meters.*

The spread in outlet temperatures was gradually improved by continued cold trapping to remove oxide, by removing some fuel elements and washing them down, and by "jiggling" the fuel elements in their process tubes. The improvement was sufficient by Dec. 28 to allow operation at full power [20 Mw(t)] but with an abnormally high total coolant flow rate (1460 gal/min rather than 1100 gal/min). Operation at various power levels continued until Jan. 29, the scheduled end of the run. On Jan. 7 naphthalene had been detected in the reactor cover gas, indicating that Tetralin had entered the primary sodium system at some earlier time. A Tetralin leak had occurred in the bearing housing casting of the main primary pump in June 1958, but it was not known whether any Tetralin had leaked into the primary system.

Operation of the reactor continued during the period from Feb. 14, 1959, to May 24, 1959, according to the planned schedule of runs and tests. Some further difficulties with the exit coolant temperature spread were experienced, but not enough to prevent operation at full power; and, on the average, these difficulties continued to decrease. The last run of this series, which was made from May 14 to May 24, was a high-temperature run, in which the sodium outlet temperature was raised to 1065°F for about 1 hr, at a reactor power of 6 Mw(t) and a sodium flow rate of 500 gal/min. A number of fuel elements were inspected during the series of runs, and no unusual deterioration was observed.

During the succeeding run (Run 13, May 27 to June 3), at full power with 1000°F coolant outlet temperature, unusual behavior was observed, beginning on May 30:

1. The reactor inlet temperature rose slowly from 545 to 580°F over a period of about three days.

2. The log mean temperature difference across the intermediate heat exchanger began to increase, indicating an impairing of its heat-transfer characteristics.

* The plugging meters consist of perforated plates in the sodium coolant system, installed in bypass lines where the sodium can be cooled to determine the temperature at which sodium oxide precipitates and plugs the perforations.

3. A thermocouple located in a fuel slug in one of the elements showed an increase from 860 to 945°F, over a period of about 20 min. A similar thermocouple in another fuel element did not show a corresponding increase.

4. Some of the fuel-channel exit temperatures showed a slight temperature increase of about 10°F.

5. The temperature rise of the moderator coolant fluctuated abnormally, by about 18°F, for a period of about 4 hr, and then jumped abruptly about 30°F.

6. One of the temperature probes in a coolant channel showed fluctuations of about 30°F.

7. The reactivity increased gradually by about 0.3 per cent over a period of about 6 hr. A further, slower increase of about 0.1 per cent occurred over the next three days of operation.

Near the end of the run a Tetralin leak was discovered in the thermocouple well in the freeze seal of the main circulating pump. The run was terminated to repair the pump. Seventeen fuel elements were examined visually (by television) after the run and were found to be slightly dirty but in good condition. Nitrogen gas was bubbled through the primary sodium to remove Tetralin, from June 17 to July 5. The nitrogen used amounted to 400,000 cu ft, and about three pints of Tetralin and 1500 cm^3 of naphthalene crystals were recovered. The sodium temperature was varied from 350 to 425°F during this process. It is stated that little decomposition of Tetralin occurs at temperatures below 475°F, that in the temperature range 470 to 675°F decomposition is rather slow, yielding naphthalene and hydrogen as the major decomposition products, and that, at the SRE design sodium temperature, decomposition is rapid: to hydrogen, free carbon, and a mixture of low-molecular-weight aliphatic hydrocarbons. It has not been possible to make a precise estimate of the total amount of Tetralin which may have leaked into the sodium system. It is believed that the amount lies between 1 and 10 gal.

After the primary pump had been reinstalled with a NaK-cooled freeze seal, a new run was begun (Run 14, July 14 to July 26). There were again large differences among fuel-channel outlet temperatures and abnormal fluctuations of the temperature rise of the moderator coolant. An abnormally high radioactivity level was observed in the reactor room on the afternoon of July 12, and a temporary high level (1.5×10^{-4} $\mu c/cm^3$) occurred in the stack gas activity. The reactor was shut down, and a sodium-level indicator thimble, which was suspected of leaking cover gas and thereby causing the high activity level, was removed and replaced by a standard plug. It was concluded later that the leak in the thimble had existed for some time and that the sudden increase in room activity was probably due to fission products escaping into the cover gas from a failed fuel element or elements. After restart of the reactor it was observed that the moderator coolant temperature rise did not respond properly to an increase in the total coolant flow; it appeared that very little sodium was leaking across the core grid plate to cool the moderator cans in the normal fashion. As the reactor power level was brought up, the power appeared not to respond properly to control-rod motion. Finally the reactor attained a positive period of $7\frac{1}{2}$ sec and was scrammed manually; the peak power indicated was about 24 Mw(t). After restart the reactor continued to show wide divergences of outlet coolant temperatures, a build-up of the radioactivity level in the reactor, and an indicated high fuel temperature in an experimental fuel element. Despite the short-term increase in reactivity that caused the power excursion, the records of critical control-rod position show that the reactor, on the average, was gradually losing reactivity; the net loss over the first four days of the run amounted to about 1.2 per cent k_{eff}. Operation at low power was continued, with interruptions, for several days. On July 26 the reactor was shut down and fuel elements were examined. The examination revealed fuel-element damage.

Subsequent examination has shown that of the 43 fuel elements* in the SRE, 10 were parted near the middle so that the lower ends of the elements remained in the core when the upper ends were lifted out, one fuel element had damaged jackets, and two elements, at the time of

* A fuel element consists of a cluster of seven rods of full core length. Each rod consists of a 6-ft-high column of uranium metal slugs, 0.75 in. in diameter and 6 in. long, in a stainless-steel jacket tube of 0.010-in. wall thickness. The slugs are thermally bonded to the jacket by a 0.010-in. NaK annulus. Outside rods are spaced by spirally wrapped stainless-steel wire, and the cluster is normally supported by hanging from the top.

writing of the report, were still stuck in the core. The locations of the damaged elements were distributed over the core, not strongly concentrated in the regions of highest power density. Two of the damaged elements had contained thermocouples to read fuel center temperature. Of these, one had indicated a maximum temperature of 1465°F on July 23, and it was felt that the actual maximum might have been at a different point, and considerably higher. The other recorded a maximum temperature of 1100°F on July 24. The temperature data are confusing since there were periods in both cases during which temperatures were not recorded; also the maximum readings occurred after the most probable period of fuel-element failure. The design central temperature of the fuel elements is 1200°F; the melting point of the uranium-iron eutectic is 1340°F. In addition to the fuel-element damage, some damage to the zirconium moderator cans was considered probable, and damage to the intermediate heat exchanger was considered possible. A later report[5] has indicated that at least three of the moderator cans require replacement and that more may have to be replaced because the fuel from the lower ends of the parted fuel elements cannot be extracted without damaging them.

The conclusions reached in the interim report were as follows:

1. The fuel cladding was damaged as a result of the formation of low-melting iron-uranium alloy. The diffusion of uranium into the stainless steel to form the alloy was greatly accelerated by local high temperatures which resulted from partial blockage or fouling of the heat-transfer surfaces of the coolant passages. The actual mechanism of channel blockage is not understood, but it can, with reasonable certainty, be attributed to the decomposition products of Tetralin. Possibly the presence of low concentrations of sodium oxide and sodium hydride in the coolant contributed to the fouling or blockage.

2. It seems quite likely that the first cladding damage occurred during the afternoon of July 12, just before the high radioactivity level was observed in the reactor room. This was before the occurrence of the positive reactivity excursion. It is possible, although far from certain, that no fuel elements parted after July 16.

3. The high-temperature runs which immediately preceded the run during which the damage occurred bear no known relation to the fuel-element cladding damage.

4. General explanations have been postulated for the reactivity changes observed during the final run, but no definite conclusions can yet be drawn. The gradual loss of reactivity, which amounted to 1.2 per cent over the first four days of the run, might be attributed to damage to a few zirconium process tubes and subsequent leakage of sodium into the graphite, or to the dropping of the lower sections of 10 or 12 fuel elements to a region of lower reactivity worth,* or to a combination of these effects. If three or four moderator logs were completely saturated, the reactivity loss would be about 1.2 per cent; the reactivity loss from parting and dropping of the lower section of a single fuel element would be less than 0.1 per cent. No rational explanation of the cause of the more rapid positive excursion, of about 0.3 per cent k_{eff}, is as yet available. Further examination of the core may provide this answer.

5. Indications are that the piping and components of the primary coolant system are undamaged (by carburization or nitriding) with the possible exception of the intermediate heat exchanger, which may be more susceptible because of the relatively thin (0.058 in.) walls of the tubes. The sodium contains a considerable amount of carbon. The fission-product contamination of the coolant does not appear to be a major concern even in the remote event of a major sodium fire. The apparently undamaged fuel elements from the reactor are suspect because of unknown local temperature histories. The moderator cans are suspect because of possible rupture.

6. Despite the major fuel-element damage and the subsequent release of fission products to the coolant and cover gas, no hazard was presented to the reactor environment. No iodine was detected in the cover gas; it appears that all the iodine released from the fuel was retained by the coolant. It appears also that the oxide cold-trap system may have removed some fission products from the sodium.

As pointed out in the above conclusions, some of the results of the occurrence are encouraging

* The maximum drop possible is $10\frac{3}{4}$ in.

with regard to the general behavior of metal-fuel sodium graphite reactors under conditions of gross fuel-element failure. The fact that approximately one-fourth of the fuel elements could fail without causing a major contamination problem or any external hazard is encouraging. At the same time the history of the accident does illustrate the difficulty of determining when potentially damaging conditions exist in the reactor.

Apparently the primary cause of the trouble (the Tetralin leak) occurred at least as early as May 30, and the existence of the leak was established by June 3. The interval from June 3 to July 26 was occupied with unsuccessful attempts to clean up the system and restore normal operation. In reading the account of these operations and the operations prior to discovery of the leak, one is struck by the evident difficulty of assessing the condition of the internal reactor parts. A factor contributing to this difficulty appears to be the sodium oxide effect. Apparently it was often difficult to tell whether abnormal coolant-flow conditions and abnormal temperatures were due to oxide or to other contaminants.

It further appears that there was difficulty in recognizing, by the presence of fission products in the coolant or the cover gas, the occurrence of fuel-element failure. The reason for this difficulty is not readily apparent from the report.

It is important that the causes of the reactivity changes during the final run be determined. Reactivity changes are not surprising during a period in which damage to fuel elements, and possibly to moderator cans, is occurring. It is relatively easy to postulate causes of reactivity loss. The most obvious possibilities are the dropping of the bottom sections of the parted fuel elements and the leakage of sodium into the moderator cans. The magnitude of the total reactivity loss (about 1.2 per cent k_{eff}) indicates that the latter process was at least partly responsible. The processes by which reactivity may be gained, or by which reactivity previously lost may be partially regained, are not so obvious. The power excursion analysis,[3] which is not yet completed, indicates than an unexplained reactivity increase of some 0.3 per cent k_{eff} occurred over a period of about 150 sec just before the reactor was scrammed. Although it is conceivable that the positive moderator temperature coefficient of reactivity could account for the reactivity gain, a cause for such a rapid moderator temperature increase is not obvious. (The moderator temperature coefficient of reactivity is $+1.7 \times 10^{-5}/°F$. The Doppler coefficient of reactivity, which applies to the fuel temperature, is $-1.1 \times 10^{-5}/°F$.[3])

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MOLTEN PLUTONIUM REACTORS:
THE FAST REACTOR CORE TEST FACILITY

The Los Alamos Scientific Laboratory (LASL) has been conducting research and development programs aimed toward a solution of the problems connected with the plutonium power-reactor concept. One of the more recent "hardware" type developments has been the construction of LAMPRE-I, a molten-plutonium-fueled experimental reactor. The Laboratory has also proposed construction of the next step after LAMPRE-I, the Fast Reactor Core Test Facility (CTF).

The LAMPRE-I has the following essential features:[1-3]

Fuel alloy:	molten plutonium – 9.5 at.% iron
Container:	304 stainless steel
Reflector:	Armco iron and stainless steel (no breeding envisioned)
Coolant:	sodium
Power:	1 Mw(t) (dumped via sodium-air exchanger)

The core tank consists of 199 capsules with the fuel alloy contained in 143 of the tantalum capsules and the coolant flowing externally. Most of the remaining pins will be unfueled stainless steel and serve as a radial reflector. The core tank is positioned between the upper and lower reflectors with control affected by means of movable (in a vertical direction) reflector type mechanisms. The LAMPRE-I is in the final assembly stage, and the hazards report has been issued.[3] One of the major objectives of the development program is the demonstration of the feasibility of the long-term containment of liquid plutonium alloy, and the LAMPRE-I itself is an essential step in the container development program. It does not, however, duplicate the neutron flux and the thermal flux to be expected in a molten-fuel power reactor; hence the need for the CTF. The latter facility is expected to yield the remaining essential information to evaluate correctly the molten-fuel reactor concept from the technical and economic standpoints.

Objectives

The detailed objectives of the CTF program, as stated in reference 4, are summarized below:

Fission-gas Purging. At the very high power densities expected in molten-fuel reactors, the gaseous fission products are generated at a relatively rapid rate. For a core operating at 500 Mw/ton and under 4 atm pressure, the gases generated in 1 hr would occupy about 1 per cent of the core volume. Gas-purging methods would be tested in the CTF in an environment similar to full-scale power reactors.

Specific Power. The specific powers of economic interest pose questions as to thermal-stress fatigue, fabrication and testing of finely divided heat-transfer surfaces, structural stability of core, gas purging in small passages, and other operational problems.

In-pile Processing. One of the features of the molten-fuel core is the possibility of continuous fuel processing. It is intended that at some later date a core will be tested employing some sort of in-pile processing loop for the removal of nongaseous fission products.

Blanket Development. Since it appears that the molten-plutonium reactor cores will be characterized by high fast-neutron leakage, the blanket design must be tailored to take advantage of the large direct production of fission power in U^{238}. In addition, there has been little effort toward low-cost blanket-plutonium recovery, an area where significant cost reductions may be possible. Breeder blanket development keyed to economic optimization will be an important objective of the CTF after successful demonstration of a core design.

Control Methods. Very sizable prompt negative coefficients of reactivity are characteristic of the molten-plutonium cores presently considered. This arises, in part, from the fact that the fuel alloy has a large volume coefficient of

thermal expansion. Although this is an advantage with respect to safety considerations, it means that large amounts of reactivity control must be incorporated into the system, and this may be a difficult mechanical problem. In addition, little is known as to the susceptibility of molten cores to bowing-induced instabilities.

Breeding Ratio. The existing experimental data for predicting breeding ratios of large plutonium reactors is inadequate. Breeding ratio measurements would be made on the CTF.

Developmental Testing. The CTF may be used for various types of developmental tests, including tests of core materials under conditions approaching those in large molten-fuel reactors, tests of complete cores, and tests of prototype steam generators.

Design Data

A summary of CTF design and performance data is presented[4] in Table XI-1. Since the facility is designed as one to test cores, provisions have been made to replace the reactor core, upper and lower axial blankets, fuel-circulation system, and shield plug with a minimum of difficulty. This is accomplished by canning these components in a "core capsule" about 10 in. in diameter by 25 ft in length. This capsule could be removed and replaced by more or less conventional refueling machinery.

Several possible core structures are being investigated, and the most promising appear to be (1) a core fabricated of flat spiral coils of tubing stacked one above the other and connected to form a continuous fuel channel or (2) a spirally wound sheet structure wherein the channel is formed by welding spacer strips between two sheets. A fuel reservoir would then be connected to this structure, so designed that the fuel would be maintained in a subcritical condition while in the reservoir; the fuel would be displaced into the core by means of the addition of sodium to the reservoir. To remove fission gases, the fuel would be circulated out of the core by means of a sodium lift pump to a region where disengagement would occur; no attempt is to be made, at present, to remove nongaseous fission products in the CTF operations, but the reprocessing of the molten fuel is being studied on a laboratory scale for possible future application to the CTF.

Table XI-1 A SUMMARY OF DESIGN AND PERFORMANCE DATA[4]

Core Data	
Power	15,000 kw
Dimensions	20.4 cm OD by 20.4 cm long (8 by 8 in.)
Volume	5.91 liters
Composition:	
Pu-Fe alloy	30.6 vol.%
Container materials	20.9 vol.%
Coolant (sodium)	48.5 vol.%
Fuel alloy	90.5 at.% Pu, 9.5 at.% Fe 97.6 wt.% Pu, 2.4 wt.% Fe
Heat-transfer surface	20.7 sq ft
Average heat flux	2.53×10^6 Btu/(hr)(sq ft)(°F)
Power density	8280 kw/liter of fuel, 2540 kw/liter of core
Specific power	517 watts per gram of plutonium
Core sodium flow rate	6.8×10^5 lb/hr
Coolant flow area	0.155 sq ft
Sodium velocity	23.5 ft/sec
Average temperature rise in coolant	140°C (250°F)
Average fuel temperature	~650°C (1200°F)
Maximum coolant temperature	650°C (1200°F)
Maximum fuel temperature	~760°C (1400°F)
Blanket Data	
Blanket power	5000 kw
Axial blanket dimensions (each)	25 cm in diameter by 50 cm long (10 in. in diameter by 20 in. long)
Control blanket dimensions	60 cm OD by 27 cm ID by 120 cm long (23 by 10.5 by 48 in.)
Radial blanket dimensions	118 cm OD by 62 cm ID by 120 cm long (47 by 23 by 48 in.)
Composition:	
Axial and control blankets	65 vol.% U–Mo alloy 35 vol.% Na and Fe
Radial blanket	80 vol.% U–Mo alloy 20 vol.% Na and Fe
Primary Coolant System	
Coolant	Sodium
Heat-removal capacity	20 Mw
System sodium flow rate	910,000 lb/hr
Pump type	Centrifugal, variable speed
Sodium temperatures:	
IHX sodium inlet (shell side)	590°C (1100°F)
IHX sodium outlet (shell side)	450°C (850°F)
Secondary Coolant System	
Coolant	Sodium
Heat-removal capacity	20 Mw
System sodium flow rate	910,000 lb/hr
Pump type	Centrifugal, variable speed
Maximum pump capacity at 290°C (550°F)	2300 gal/min

Table XI-1 (Continued)

Sodium temperatures:	
IHX inlet (tube side)	400°C (750°F)
IHX outlet (tube side)	538°C (1000°F)
Superheater inlet	538°C (1000°F)
Superheater outlet	503°C (938°F)
Tempering heater inlet (shell side)	503°C (938°F)
Tempering heater outlet (shell side)	390°C (733°F)
Tempering heater inlet (tube side)	288°C (550°F)
Tempering heater outlet (tube side)	400°C (750°F)
Evaporator inlet	390°C (733°F)
Evaporator outlet	288°C (550°F)
Steam Generator	
Design capacity	20 Mw
Steam flow rate	62,500 lb/hr
Outlet steam pressure	900 psig
Steam and water temperatures:	
Superheater outlet steam	496°C (925°F)
Evaporator outlet steam	280°C (535°F)
Evaporator inlet water	204°C (400°F)
Feed-water quality:	
Total solids	<1 ppm
*p*H	9.5 to 10.0
Construction materials:	
Superheater	316 stainless steel
Evaporator	2.25% Cr–1% Mo steel
Tempering sodium heater	2.25% Cr–1% Mo steel

Since a study of breeding is proposed as one of the objectives of the CTF, fertile blankets are incorporated into the design. The fixed radial blanket is composed of alloy pins and is cooled by the inlet sodium. The control blanket occupies space between the fixed radial blanket and the core and is designed so that the lower two-thirds of the cylinder can be lowered vertically about 1 ft in several independently operated segments, thereby "uncovering" the core in a radial direction. The control blanket and the core-axial blanket combination are cooled by sodium in parallel flow.

The facility contains a steam generator thermally connected to the primary coolant by means of the secondary sodium system. The steam generator is of the once-through, nonrecirculation type with no steam or water drum. It consists of three sections: an evaporator, a sodium-to-sodium tempering heater, and a superheater. The shells of all three sections are welded in series to form a continuous U-bend unit. The secondary sodium flows through the shell, and the water and steam of the evaporator and superheater sections flow within the tubes. The outlet sodium from the shell (i.e., the low-temperature outlet from the evaporator section) is run through the tubes of the tempering heat exchanger, which is installed between the superheater and evaporator to reduce the temperature of the shell-side sodium from 940°F at the superheater outlet to 730°F at the evaporator inlet. This temperature reduction was found necessary to achieve acceptable thermal stress in the tubes of the evaporator. Both primary and secondary coolant loops employ centrifugal, sump type pumps with variable-speed magnetic coupling to the drive motors. The superheated steam produced is desuperheated to about 700°F, passed through a throttle simulator valve, and condensed in an air-cooled condenser.

As a pilot plant to the CTF steam generator, a 2-Mw steam generator test rig is being constructed at Los Alamos. Although no details are given in reference 4, the installation apparently consists of a gas-fired primary sodium loop coupled to a secondary sodium loop through the secondary heat exchanger. Sump type centrifugal pumps are utilized, and the heat is dumped in a 2-Mw steam generator. This test is scheduled for operation sometime in early 1960, and, if the designs prove satisfactory, the steam generator for the CTF will be a scaled-up version of the 2-Mw unit. The basic approach to obtaining a reliable steam generator is the use of a simple design and the application of rigorous quality-control measures during construction. The steam generator employs thoroughly inspected single-wall tubing; thermal expansion in tubes and shells is accommodated through U-tube construction, and all tube sheets are insulated from the sodium by a gas space. In addition, a method has been developed to form a crevice-free joint between tubes and tube sheets, such a joint being readily inspected by radiographic techniques. The reference states that the "... experimental unit ... is believed to represent a distinct advance over previous generators in assured integrity, conservative stress design, and efficiency and flexibility of operation."[4]

It is proposed that construction of the CTF be started in January 1961 and that the first core be installed in January 1963. LAMPRE-I is scheduled for initial operation in January 1960, and it is estimated that 18 months of operation would provide sufficient data, in addition to other laboratory programs, for the design of the first

core for the CTF. The cost of the CTF (not including the concurrent core-development program) is estimated[4] as $6,826,000.

Status of Core Development

Reference 4 contains a summary of the current status of the molten-plutonium core development. The core accounts for most of the problems of the reactor type other than those related to the handling and technology of the highly radioactive, highly toxic fuel. Problems of the latter type were not discussed, but presumably they are severe.

A major core problem is to find a suitable plutonium alloy which has a sufficiently low melting point and which provides the proper degree of dilution of the plutonium. Some dilution is evidently needed to provide sufficiently high heat-transport capability per unit of plutonium, but the permissible degree of dilution is limited by considerations of criticality and neutron losses. The plutonium-iron eutectic (9.5 at.% iron; m.p. 410°C) scheduled for use in LAMPRE-I is considered suitable for this small reactor. The properties of the eutectic are summarized in the reference. For larger reactors, higher dilution is needed. Although cobalt and nickel also form plutonium eutectics with melting points below 500°C, their concentrations in the eutectics are also small (12 at.% cobalt; 14 at.% nickel). There are no known low-melting binary eutectics of plutonium which contain less than 86 at.% plutonium, and it appears that the only possibility for diluting the plutonium to less than about 80 at.% plutonium without markedly increasing the melting temperature of the liquid fuel is to add a third element. Cerium is a promising third element. In the ternary systems employing cerium, the lowest-melting-point alloys are usually those in which cerium is the major constituent. Reference 4 states: "Thus it appears that in the three ternary systems, Pu-Ce-Co, Pu-Ce-Ni, and Pu-Ce-Cu, alloy compositions exist that have melting temperatures low enough for these alloys to be used as liquid fuels for reactors provided the plutonium content does not exceed 40 at.%."

The program for developing a container material for the liquid fuel has been primarily concerned with tantalum and its alloys. Tungsten is less subject to attack by plutonium, but its fabrication problems have barred it from serious consideration. Yttrium has recently shown promise. Intergranular penetration and transgranular attack or mass transfer have been observed on tantalum. The solution type of attack can be reduced to negligible values at temperatures below 800°C by the rigid exclusion of oxygen from the fuel and the addition of small amounts of dissolved carbon. The grain boundary attack is believed to result when impurities precipitate interstitially in sufficient quantity to form a continuous path for the attacking plutonium. High-purity tantalum has withstood 1500-hr exposure to plutonium fuel at 650°C with no sign of attack. This material was produced by starting with an arc-cast material having a total content of interstitial elements (hydrogen, oxygen, nitrogen, and carbon) of less than 100 ppm. The starting material was remelted by an electron beam in high vacuum to achieve an oxygen content of 15 ppm or less; the material was then rolled into sheet stock with rigid control of oxygen pickup during anneals. It is hoped that the precipitation of tantalum oxide in grain boundaries can be prevented by the addition of an alloying agent which would preferentially acquire the oxygen and either keep it in solution or cause it to precipitate in discontinuous granules. Limited, but encouraging, results have been had with tungsten, rhenium, and yttrium as alloying additions.

High-purity tantalum is readily workable by most normal means of fabrication, but extreme care must be used during fabrication to maintain its purity. Although it is easily weldable by the proper technique, the effects of welds on corrosion resistance have not been completely evaluated. In LAMPRE-I the welding problem has been bypassed by deep-drawing the fuel containers and thus avoiding welds in contact with the fuel. The high-purity tantalum is expected to be adequate for the LAMPRE-I conditions, although one of the purposes of the experiment is to determine whether radiation effects add to the difficulties.

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1. J. A. Lane, H. G. MacPherson, and F. Maslan, *Fluid Fuel Reactors*, Addison-Wesley Publishing Company, Inc., Reading, Mass., 1958.
2. R. M. Kiehn et al., LAMPRE, A Molten Plutonium Fueled Reactor Concept, LA-2112, January 1957. (Unclassified AEC report.)
3. E. O. Swickard et al., Los Alamos Molten Pluto-

nium Reactor Experiment (LAMPRE) Hazard Report, LA-2327, Dec. 28, 1959. (Unclassified AEC report.)

4. D. B. Hall et al., A Preliminary Study of a Fast Reactor Core Test Facility, LA-2332, August 1959. (Unclassified AEC report.)

Errata

The September 1959 issue of *Power Reactor Technology* (Vol. 2, No. 4) contained a review of the large power-reactor studies carried out by the Atomic Energy Commission as authorized by Public Law 85-590 (Section 101). In a quotation from reference 1, which covered the work on organic-cooled reactors, *Power Reactor Technology* inadvertently omitted a line. The last two sentences of the quotation on page 19 of Vol. 2, No. 4, should read:

> The pyrolytic damage rates are 0.00010 wt.%/hr at 550°F, 0.0023 wt.%/hr at 675°F, and 0.0135 wt.%/hr at 745°F; hence, the total pyrolytic damage is approximately 670 lb/25 hr. From this, the total coolant make-up rate is approximately 26,570 lb/24 hr or 1105 lb/hr. Using $0.17 per pound for new coolant, the coolant make-up cost is 0.63 mill/kw-hr.

The underlined section was omitted. This omission had the effect of ascribing the total coolant make-up rate to pyrolytic damage rather than to the combination of pyrolytic and radiolytic damage.

The total coolant make-up rate of approximately 26,570 lb/24 hr was also incorrectly quoted as 26,750 lb/24 hr.

Decomposition Rates in OMRE

Mr. R. Balent, Deputy Director of the Organic Reactors Department, Atomics International, has supplied the following information on observed decomposition rates in OMRE, which supplements and clarifies the material discussed at the bottom of page 19 and the top of page 20 in *Power Reactor Technology*, Vol. 2, No. 4:

> The values of 86 and 90 lb of terphenyl decomposed per thermal megawatt-day of reactor operation represent decomposition rates observed at very low concentrations of decomposition products in the coolant (0 to 12 per cent high boiling compounds). Operation of the OMRE has shown that these decomposition rates decreased very strongly with increasing concentrations of high boiler compounds in the coolant. At 30 per cent high boiler concentrations in the coolant, the observed decomposition rate was 51 lb per thermal megawatt-day of reactor operation.
>
> The value of 125 lb of decomposition products formed per thermal megawatt-day of reactor operation was obtained at a maximum fuel-element surface hot-spot temperature of 750°F and a bulk coolant temperature of 700°F. Under these conditions the OMRE was operating at a power level of only 2.0 Mw. An appreciable fraction of the decomposition of the coolant was attributable to pyrolytic damage in the OMRE coolant system outside of the high-flux region of the reactor. This damage rate, when it is expressed as pounds per megawatt-day in a reactor operating at very low power, represents a significant increase over the radiolytic damage rate (at zero power, for instance, there would be no radiolytic damage and the pyrolytic damage rate per megawatt-day would be infinite). In a full-scale reactor operated at high power, the pyrolytic decomposition rate is practically insignificant at these temperatures.

We appreciate the opportunity to publish this information, which clarifies the OMRE results and points out their practical interpretation.

Reference

1. Bechtel Corp. and Atomics International, Organic-cooled Reactor Study, TID-8501, April 1959. Part 1, Summary of Study (previously available as AEC Report BCPI-1, Vol. 1); Part 2, 300 Mw Power Plant Conceptual Design (previously available as AEC Report BCPI-1, Vol. 2); Part 3, Reactor Concept Evaluation (previously available as AEC Report BCPI-1, Vol. 3); Part 4, 75 Mw Power Plant Conceptual Design (previously available as AEC Report BCPI-1, Vol. 4); Part 5, 300 Mw Coal-fired Power Plant Comparison Study (previously available as AEC Report BCPI-1). (Unclassified AEC report.)

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